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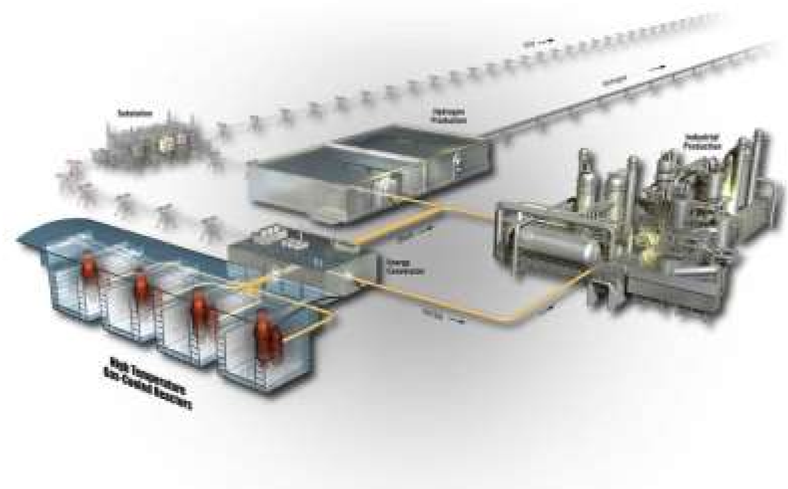
Technical Program Plan for INL Advanced Reactor Technologies Technology Development Office/ Advanced Gas Reactor Fuel Development and Qualification Program



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TECHNICAL PROGRAM PLAN FOR INL ADVANCED REACTOR TECHNOLOGIES TECHNOLOGY DEVELOPMENT OFFICE/ ADVANCED GAS REACTOR FUEL DEVELOPMENT AND QUALIFICATION PROGRAM	Identifier: PLN-3636 Revision: 6 Effective Date: 06/28/2017	Page: i of **xi**

INL ART TDO PROGRAM	Plan	eCR Number: 650079

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REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	09/30/10	All	New issue, refer to those below for earlier versions: INL/EXT-05-00465, Revision 2, July 2008 INL/EXT-05-00465, Revision 1, August 2005
1	12/3/2012	All	Updated to incorporate changes to project planning and complete biennial review of document.
2	12/18/2012	4, 8, 9, 42, 43, 44, 51, 52, 53, 56, 57, and 64	Revised to incorporate INL Regulatory Affairs and NRC agreements and commitments.
3	05/05/2014	All	Revised/updated per routine review.
4	05/07/2015	All	Revised/updated per routine review.
5	05/09/2016	All	Revised/updated per routine programmatic review.
6	06/28/2017	All	Revised/updated per routine review.

SUMMARY

High-temperature gas-cooled reactors (HTGRs) are graphite-moderated nuclear reactors cooled with helium. Their high outlet temperatures and thermal-energy conversion efficiency enable efficient and cost-effective integration with non-electricity-generating applications. These applications include process heat and hydrogen production for petrochemical and other industrial processes that require operating temperatures between 300 and 900°C. HTGRs will supplement the use of premium fossil fuels such as oil and natural gas, improve overall energy security in the United States by reducing dependence on foreign fuels, and reduce carbon dioxide (CO_2)/greenhouse gas emissions. The HTGR design uses helium as a coolant, graphite as a neutron moderator, and ceramic particle fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons, retains its strength at high temperatures, provides structural stability, and acts as a substantial heat sink during transient conditions. The ceramic particle fuel is extremely robust and retains the radioactive by-products of the fission reaction within the coated particle under normal and off-normal conditions.

The United States Department of Energy (DOE) Office of Nuclear Energy and the Idaho National Laboratory (INL) Advanced Reactor Technologies (ART) Technology Development Office (TDO) Advanced Gas Reactor (AGR) Fuel Development and Qualification program (referred to as AGR Fuel program hereafter) are pursuing qualification of tristructural-isotropic (TRISO) particle fuel for use in HTGRs. The AGR Fuel program was established to achieve the following overall goals:

- Provide a fuel qualification data set in support of the licensing and operation of an HTGR. HTGR fuel performance demonstration and qualification comprise the longest duration research and development (R&D) tasks required for design and licensing. The fuel form is to be demonstrated and qualified for service conditions that include normal operation and potential accident scenarios.
- Support deployment of HTGRs for hydrogen, process heat, and energy production in the United States by reducing market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Extend the value of DOE Office of Nuclear Energy resources by using international collaboration mechanisms where practical.

TRISO particle fuel development and qualification activities support prismatic and pebble-bed HTGR fuel designs. The AGR Fuel program to date has focused on manufacturing and testing the fuel design for HTGR concepts using the most recent gas-turbine modular-helium reactor fuel product specification[a] as a starting point. Irradiation, safety testing, and post-irradiation examination (PIE) plans will support fuel development and qualification in an integrated manner. Preliminary operating conditions and performance requirements for the fuel and preliminary fuel product specifications to guide the AGR Fuel program's fuel fabrication process development activities are based on previously completed HTGR design and technology development activities, operating conditions, and performance requirements.

At the onset of the AGR Fuel program in 2002 (then known as the Very High Temperature Reactor [VHTR] TDO/AGR Fuel program), facilities and personnel experienced in activities necessary to address the program goals existed in the United States, primarily at INL and Oak Ridge National Laboratory (ORNL). INL and ORNL personnel with experience and knowledge of TRISO particle fuel, facility

a. DOE-GT-MHR-100209, "Fuel Product Specification," May 1994.

status, and capabilities were involved in developing the initial Technical Program Plan for the Next Generation Nuclear Plant (NGNP)/Advanced Gas Reactor Fuel Development and Qualification Program.b In addition, General Atomics provided input regarding prismatic HTGR fuel performance requirements and perspectives from its experience in fuel development, fuel fabrication, and fuel-related analytical capabilities needed to support licensing interactions. BWX Technologies Inc. also provided input based on its experience and capabilities for fuel-kernel production and fuel-particle coating. Many of the individuals who helped develop this plan were directly involved in producing and testing previous U.S. fuel for the modular high-temperature gas-cooled reactor and the new production reactor, and they conducted extensive investigations and reviews in the early 1990s following the unexpectedly high fuel failure levels observed in those tests. This plan builds directly on the large body of coated-particle fuel experience and is generally consistent with the recommendations arising from those experiences.

Based on the recommendation of the Nuclear Energy Advisory Committee to Congress in 2011, design-specific efforts on the NGNP project were halted at the end of the conceptual design phase in 2011, in part because a viable public-private partnership for a demonstration plant and follow-on commercialization had not yet been established. With no HTGR reactor deployment anticipated in the near term, the R&D program focus is to qualify a fuel and establish a commercial fuel vendor in the United States. There has not been, and there will not be, an effort to verify or validate any potential reactor vendor codes as a part of the HTGR R&D performed under the AGR Fuel program. The effort to quantify fission product transport within reactor core materials and provide a technical basis for the source term has similarly been halted after initial hydrogen and tritium permeation testing in various stainless-steel alloys.

The AGR Fuel program involves five major program elements:

1. *Fuel Fabrication*. This program element, to fabricate successful TRISO particle fuel (manufacturing fuel that meets the fuel quality and performance requirements for licensing an HTGR), requires developing a coating process that replicates the HTGR particle design, to the greatest extent possible, properties of the coatings on German fuel particles that have previously exhibited superior irradiation and accident performance. Coating-process development has been accomplished in two phases: initially in a 2-in.-diameter, laboratory-scale coater (AGR-1) followed by scale-up to a 6-in., prototypic, production-scale coater (AGR-2). The Fuel Fabrication program element has included establishing the fuel-fabrication infrastructure; developing the process for the low-enriched uranium carbide/oxide kernels, TRISO particles, and compacts; developing coating process models; developing quality control methods; performing fuel process scale-up analyses; and developing process documentation for technology transfer to private industry. The fuel-fabrication effort produces TRISO particle fuel within cylindrical fuel compacts that meets fuel product specifications and provides fuel and material samples for characterization, irradiation, safety testing, and PIE as necessary to meet the overall AGR Fuel program goals.

[b] ORNL, *Technical Program Plan for the Advanced Gas Reactor Fuel Development and Qualification Program*, ORNL/TM-2002/262, April 2003.

2. *Fuel and Material Irradiation*. This program element provides data on fuel performance during irradiation to support fuel process development, qualify a fuel design and fabrication process for normal operating conditions, and support development and validation of fuel performance and fission product transport models and codes. This program element also provides irradiated fuel and materials necessary for PIE and safety testing. Seven irradiation tests, designated as AGR-1 through AGR-7, have been defined to provide data and sample materials within the AGR Fuel program.

3. *Fuel PIE and Safety Testing*. This program element provides the facilities and processes to measure the performance of TRISO particle fuel under normal operating and potential accident conditions. Moisture and air ingress testing in quantities expected to exist within the typical helium and neon gas supplies used during irradiation (testing performed during AGR-3/4 irradiation) and safety testing (planned to be performed during AGR-5/6/7 PIE) will be performed to determine their effects on TRISO particle fuel. This work supports the fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts during irradiation and under potential accident conditions. PIE and safety testing provide a broad range of data on fuel performance and fission product transport within TRISO-coated fuel particles, compacts, and graphite materials representative of fuel element blocks. These data, in combination with the in-reactor measurements (irradiation conditions and fission gas release-rate-to-birth-rate ratios), are necessary to demonstrate compliance with fuel performance requirements and support developing and validating computer codes.

4. *Fuel Performance Modeling*. This program element addresses the structural, thermal, and chemical processes that can lead to TRISO-coated particle failures. It considers the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Fission product release from the fuel particles and transport in the fuel compact matrix and fuel element graphite during irradiation are also modeled. Computer codes and models will be further developed and validated as necessary to support fuel-fabrication process development.

5. *Fission Product Transport and Source Term*. This program element addresses the transport within reactor core materials of fission products produced in the TRISO particle fuel and is intended to provide a technical basis for source terms for HTGRs under normal irradiation and potential accident conditions. However, most of this work scope has not been performed because of funding shortfalls and higher priority work scope. Some initial fission product transport studies were performed on hydrogen and tritium permeation through high nickel superalloys with results that were included in published reports. An evaluation of data from irradiation and safety testing of "designed-to-fail" fuel particles will be performed as part of the AGR-3/4 post-irradiation examination. The purpose of the evaluation is to characterize fission product release and transport from TRISO particle fuel into fuel compact matrix and fuel element graphite under normal and off-normal HTGR conditions.

This plan aims to develop an understanding of the relationships among the fuel fabrication process, fuel product properties, and irradiation and safety test performance. Precise process control and advanced characterization and data-acquisition methods, conducted within a structured quality assurance framework, are important elements of achieving this objective. Producing qualified fuel performance data under fuel-irradiation conditions and in-pile gaseous fission product release, as well as a wide range of data produced during PIE and safety testing, are important elements. Fuel performance modeling is also included. The fuel performance models are considered essential for reasons, including (1) guidance for a future plant designer/applicant in establishing the reactor core design and operating limits and (2) demonstrating to the licensing authority, the Nuclear Regulatory Commission (NRC), that the applicant has a thorough understanding of the in-service behavior of the fuel system.

The five program elements and the activities associated with each are discussed in Section 3 of this technical program plan. Early AGR Fuel program activities were centered on the fuel fabrication element, because the production of fuel and materials for irradiation, safety testing, and PIE were the early critical-path activities. Now the critical-path activity is the performance of PIE and safety testing for each of the remaining experiments.

Key accomplishments of the AGR Fuel program to date are listed below:

- Developed low-enriched uranium carbide/oxide TRISO fuel fabrication and modern quality control capabilities, first at laboratory scale and then at pilot -scale at a domestic vendor facility including:
    - Improved kernel forming, carbothermic reduction, and sintering chemistries
    - Thirty-fold increase in TRISO particle capacity in the coating furnace
    - Improved methods of producing and applying resinated graphite powder overcoats to TRISO particles that eliminate multiple process steps, eliminates Resource Conservation and Recovery Act (RCRA) mixed-hazardous waste generation, and reduces production time by an order of magnitude
    - Demonstrated a multi-cavity, fully automated compacting press
    - Demonstrated a combined-cycle thermal treatment process for finishing compacts
- Developed test train designs for multi-capsule individual and multi-experiment tests.
- Completed irradiation of the first three AGR experiments: AGR-1 for 620 effective full power days (EFPDs) with no particle failures; AGR-2 for 559 EFPDs with no apparent particle failures; and AGR-3/4 for 369 EFPDs containing designed-to-fail TRISO particles that failed during irradiation.
- Completed PIE and safety testing of the AGR-1 irradiated fuel and components, including 19 safety tests through September 30, 2016, at temperatures of 1600, 1700, or 1800°C for approximately 300 hours each. In one of these tests, three compacts were simultaneously tested in a varying temperature profile (minimum temperature 830°C, maximum temperature 1690°C) simulating a temperature transient during a core-conduction cool-down event.
- Completed advanced electron microscopy and micro-analysis of as-fabricated, irradiated, and post-irradiation safety-tested AGR-1 particles at INL.
- Completed disassembly, metrology, and gamma scanning of the AGR-2 test train and its six capsules containing 66 compacts at INL.
- Completed PIE of five as-irradiated AGR-2 compacts at ORNL (four UCO compacts and one UO_2 compact).
- Completed four 1600°C safety tests and post-safety test destructive PIE at ORNL of four irradiated AGR-2 compacts (two UO_2 and two UCO). Completed one 1800°C safety test and post-safety test destructive PIE of one AGR-2 UCO compact.

- Initiated re-irradiation and safety testing of irradiated loose particles at INL. This enables study of I-131 release from AGR fuels.
- Developed equipment and methods for destructive PIE at INL of AGR-3/4 graphite rings containing fission products.
- Installed necessary equipment in radiation hot cell and initiated radial deconsolidation of irradiated AGR-3/4 compacts at INL.
- Completed conceptual design review of furnace and associated systems for safety testing irradiated fuels under air/moisture-ingress conditions at INL.
- Developed the fuel performance modeling code, PARticle Fuel ModEl (PARFUME), which has been used for irradiation pre-test and safety test predictions and refined based on information from AGR-1 PIE.
- Completed initial hydrogen and tritium fission product transport permeation of stainless-steel alloy studies.
- Established the Nuclear Data Management and Analysis System database for collection and management of data obtained during fuel fabrication, irradiation, PIE, and safety testing.
- Collected, analyzed, and qualified millions of data points generated during fuel fabrication, irradiation, and PIE for future support of NRC licensing activities of TRISO particle fuel.

In addition, in 2014, the NRC staff completed its assessment of two NGNP licensing white papers titled *NGNP Fuel Qualification White Paper*[c] and *Mechanistic Source Terms White Paper*.[d] These papers described the AGR Fuel program and its approach to determining mechanistic source terms, which relied extensively on data obtained from the AGR Fuel program. The results of the NRC's assessment were documented and transmitted to DOE via a letter with two enclosures.[e] The enclosures provided feedback on key licensing issues that are closely tied to the AGR Fuel program, the approach to fuel development and qualification, and to mechanistic source terms. These significant NRC findings indicate that the AGR Fuel program is on track to meet its goal of providing a fuel qualification data set in support of the licensing and operation of an HTGR.

c. Idaho National Laboratory, *NGNP Fuel Qualification White Paper*, INL/EXT-10-17686, Rev. 0, July 21, 2010.

d. Wayne Moe, *Mechanistic Source Terms White Paper*, INL/EXT-10-17997, Rev. 0, Idaho National Laboratory, July 21, 2010.

e. Glenn M. Tracy, NRC-Office of New Reactors, to Dr. John E. Kelly, DOE-NE, "Next Generation Nuclear Plant – Assessment of Key Licensing Issues," July 17, 2014.

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ACRONYMS

AGC	Advanced Graphite Creep
AGR	Advanced Gas Reactor
ART	Advanced Reactor Technologies
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor
BWXT	BWX Technologies Inc.
CCCTF	Core Conduction Cooldown Test Facility
DOE	U.S. Department of Energy
DTF	designed-to-fail
EFPD	effective full power day
FACS	fuel accident condition simulator
FIMA	fissions per initial heavy metal atom
FY	fiscal year
GA	General Atomics
GIF	Generation IV International Forum
HFEF	Hot Fuel Examination Facility
HFIR	High Flux Isotope Reactor
HRB	High Flux Isotope Reactor Removable Beryllium
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
IAEA	International Atomic Energy Agency
INL	Idaho National Laboratory
IPyC	inner pyrolytic carbon
LBL	leach-burn-leach
LCB	life-cycle baseline
LEU	low-enriched uranium
MFC	Materials and Fuels Complex

MHTGR	modular high-temperature gas-cooled reactor
NE	Office of Nuclear Energy
NEFT	northeast flux trap
NGNP	Next Generation Nuclear Plant
NPR	new production reactor
NQA	Nuclear Quality Assurance
NRAD	Neutron Radiography Reactor
NRC	Nuclear Regulatory Commission
OD	outside diameter
OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
PARFUME	Particle Fuel Model
PIE	post-irradiation examination
PyC	pyrolytic carbon
QA	quality assurance
QC	quality control
R&D	research and development
R/B	release-rate-to-birth-rate ratio
RCRA	Resource Conservation and Recovery Act
TC	thermocouple
TDO	Technology Development Office
TRISO	tristructural-isotropic
UCO	uranium carbide/oxide
UO_2	uranium dioxide
VHTR	very high temperature reactor
WBS	work breakdown structure

1. INTRODUCTION

High-temperature gas-cooled reactors (HTGRs) are graphite-moderated nuclear reactors cooled with helium. Their high outlet temperatures and thermal-energy conversion efficiency enable efficient, cost-effective integration with non-electricity-generation applications. These applications include process heat and/or hydrogen production for petrochemical and other industrial processes that require temperatures between 300 and 900°C. HTGRs will supplement the use of premium fossil fuels such as oil and natural gas, improve overall energy security in the United States by reducing dependence on foreign fuels, and reduce carbon dioxide (CO_2)/greenhouse gas emissions. Key characteristics of the HTGR design include using helium as a coolant, graphite as a neutron moderator, and ceramic particle fuel. Helium is chemically inert and neutronically transparent. The graphite core slows down the neutrons, retains its strength at high-temperature, provides structural stability and acts as a substantial heat sink during transient conditions. The ceramic particle fuel is extremely robust and retains the radioactive by-products of the fission reaction under normal and off-normal conditions.

The United States Department of Energy (DOE) Office of Nuclear Energy (NE) has selected the HTGR as a transformative application of nuclear energy that will demonstrate emissions-free nuclear-derived electricity, process heat, and hydrogen production. The first-of-a-kind HTGR envisioned extends past applications of gas-cooled reactor technologies and will be driven by near-term commercial industry needs and current technology availability. The reference concept will be an HTGR with a design goal outlet gas temperature of 750 to 800°C. The reactor core may be either a prismatic graphite-block core or a pebble-bed core. The reactor fuel concept will use low-enriched uranium (LEU) to obtain high burnup in a “once-through” fuel management scheme.

In developing the original version of the technical program plan, priority was given to early activities in support of near-term execution. Issues associated with longer-term activities are being addressed in more detail as they arise, and their impact is being factored into overall planning. This additional detail has not affected the basic logic of the plan but does affect the details of its execution. Based on the coordinated planning activities discussed previously, the initial technical program plan[1] was issued by Oak Ridge National Laboratory (ORNL) in April 2003. Maintaining the planning documentation was assigned to Idaho National Laboratory (INL) in 2004, consistent with its lead management role in the Advanced Reactor Technologies (ART) Technology Development Office (TDO) Advanced Gas Reactor (AGR) Fuel Development and Qualification program (hereafter referred to as AGR Fuel program). This plan has continued to be updated periodically to reflect additional knowledge and the results of ongoing and completed work. After being issued initially, the next two revisions of the plan were issued as external documents under INL document control protocol, INL/EXT-05-00465, *Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program*, Revisions 1 and 2. The documentation protocol was changed within INL in 2010 and explains the current designation as a plan document (PLN-3636). Plan execution is adjusted according to progress, results, funding changes, and limitations in terms of milestones, completion dates, and work scope. Routine revisions to the plan are issued based on the actual funding received, accomplishments, and changes in technical directions as they evolve.

1.1 Program Scope and Background

In fiscal year (FY) 2002, the DOE Office of Nuclear Energy, Science, and Technology initiated development of the AGR Fuel program for coated-particle fuel. The resulting *Technical Program Plan for Advanced Gas Reactor Fuel Development and Qualification Program*[1] and subsequent revisions defined fuel development activities to support licensing and operating an HTGR in the United States under the umbrella of the Next Generation Nuclear Plant (NGNP) project in accordance with the Energy Policy Act of 2005 when it was enacted.

Based on the recommendation of the Nuclear Energy Advisory Committee to Congress,[2] design-specific efforts on the NGNP project were halted in 2011 at the end of the conceptual design phase, in part because a viable public-private partnership for a demonstration reactor and follow-on commercialization had not yet been established. Currently, no partnership has been formed, although recently several private companies have expressed interest in using the HTGR concept in an advanced reactor design. With no HTGR deployment anticipated in the near term, the research and development (R&D) program focus is to qualify a fuel form and establish a commercial fuel vendor in the United States. The HTGR R&D will not perform verification or validation of any potential reactor vendor codes.

This latest revision of the technical program plan describes the updated path forward for developing and qualifying tristructural-isotropic (TRISO)-coated particle fuel that incorporates the experience and knowledge gained from ongoing and completed work. HTGR designs provide inherent safety, which prevents core damage under nearly all design basis accidents and hypothetical severe accidents. The principle guiding this concept is to maintain core temperatures passively below fission product release thresholds under all potential accident scenarios. The required level of fuel performance and fission product retention reduces the radioactive source term at the reactor core boundary by many orders of magnitude and, relative to the core inventory, allows potential elimination of the need for evacuation and sheltering beyond a small exclusion area. This safety approach, however, mandates exceptional fabricated fuel quality and fuel performance under normal operating and potential accident conditions. Germany produced and demonstrated high-quality fuel for their pebble-bed reactors in the 1980s, but no United States fabricated fuel had exhibited equivalent performance prior to the AGR Fuel program. As in many reactor technology development programs, fuel development and qualification were identified as essential to ensure concept viability.

A complete set of design specifications for an HTGR is not available to the AGR Fuel program, but the maximum burnup envisioned in a prismatic HTGR is within the range of 150 to 200 GWd/metric tons of heavy metal or 16.4 to 21.8% fissions per initial heavy metal atom (FIMA).[1] Maximum burnups for pebble-bed designs are typically considerably less than this. Although Germany has demonstrated excellent performance of uranium dioxide (UO_2) TRISO particle fuel up to about 10% FIMA and 1150°C, UO_2 fuel is known to have limitations because of carbon monoxide (CO) formation; kernel migration at the higher burnups; and power densities, temperatures, and temperature gradients that may be encountered in the prismatic HTGR design. With uranium carbide/oxide (UCO) fuel, the kernel composition is engineered to minimize CO formation and kernel migration, which are key threats to fuel integrity at higher burnups, temperatures, and temperature gradients. Furthermore, the performance of German silicon carbide (SiC)-based, TRISO-coated-particle, UCO fuel up to 22% FIMA (as measured by the in-pile gas release in irradiation test FRJ-P24[3]) and the excellent performance of United States made UCO fuel in AGR-1 give added confidence that high-quality SiC-based, TRISO-coated-particle, UCO fuel can be made and its superior irradiation performance statistically demonstrated.

In addition to excellent fission product retention during normal operation at high burnups and high temperatures, HTGR fuel must exhibit satisfactory fission product retention under postulated accident conditions. Limited data on the accident performance of SiC-based TRISO-coated UO_2 fuel at high burnups indicate increased cesium (Cs) releases at burnups ≥14% FIMA,[4] so safety testing is an important element. The AGR Fuel program chose to develop coated-particle fuel using a low-enriched UCO kernel to qualify a fuel to meet fuel performance requirements under specified fuel service conditions. Thus, SiC-based TRISO-coated UCO was chosen as the baseline AGR fuel to be fabricated and tested. This fuel development path complemented particle fuel development with a UO_2 kernel that was being pursued by South Africa, China, and Europe at the time. Safety testing of irradiated AGR-1 UCO TRISO compacts has demonstrated robust behavior for about 300 hours at 1600, 1700, and 1800°C, giving added confidence that SiC-based TRISO particle fuel can meet safety performance requirements.

The TRISO-coated UCO fuel specification[5] utilizing SiC as the primary fission product retention layer was developed in response to extensive evaluations[6,7] of the fuel failures experienced in irradiations in the new production reactor (NPR) and the modular high-temperature gas-cooled reactor (MHTGR) programs. This was the starting point for the fuel specification developed for the current program.[8] It is expected that this fuel will exhibit acceptable fuel performance at higher burnups (16 to 22% FIMA) at time-averaged fuel temperatures up to at least 1250°C for normal operation and 1600°C for potential accident conditions, and fast neutron fluences up to at least 5×10^{25} neutrons/m^2. This plan identifies R&D needed in the areas of fuel fabrication, fuel and materials irradiation, safety testing and post-irradiation examination (PIE), fuel performance modeling, and fission product transport and source term studies. Section 4 provides an updated integrated schedule and budget for the work required to develop, scale up to production capability, and transfer TRISO particle fuel fabrication capability to an industrial fuel vendor within the United States.

In the late 1980s, coated-particle fuel performance to the desired level of quality and predictability was demonstrated in the Arbeitsgemeinschaft Versuchsreaktor (AVR) at Jülich, Germany, and several materials test reactors. The AGR Fuel program has used a fuel design based on the most recent gas turbine modular helium reactor fuel product specification, combined with the successful German-like coating and matrix material overcoating processes. The basic structure of the AGR Fuel program is delineated in the major program elements below:

- Fuel Fabrication
    - Develop low-enriched UCO TRISO fuel fabrication and modern quality control (QC) capabilities, first at laboratory scale and then at prototypic production scale.
    - Develop capabilities for fuel fabrication at laboratory scale for establishing and refining the processing parameters.
    - Develop a modern suite of characterization and QC methods.
    - Transfer the fuel fabrication and QC technology to an industrial/commercial domestic fuel vendor.
    - Produce final reference fuel with a prototypic production-scale coater for fuel qualification testing.
- Fuels and Materials Irradiation
    - Develop test train multi-capsule designs for individual and multi-experiment tests.
    - Complete irradiation of the AGR-1 experiment for approximately 600 effective full power days (EFPDs), which is the initial shakedown test.
    - Complete irradiation of the AGR-2 experiment for approximately 550 EFPDs, which will test TRISO particle fuel made at prototypic production scale.

- Complete irradiation of the AGR-3/4 experiments for approximately 350 EFPDs that will contain designed-to-fail (DTF) TRISO particles that are expected to fail during irradiation and will provide data on fission product transport.
- Complete the irradiation of the AGR-5/6/7 experiments for approximately 500 EFPDs which will serve as fuel qualification tests and a margin test based upon the selected fuel fabrication specifications.
- Irradiation of the AGR-8 experiment originally conceived as a fission product transport validation test has been deferred at this time because of the lack of a selected reactor design, reduced funding levels, and schedule considerations.

• Safety Testing and PIE

- Perform safety testing and PIE of UCO TRISO particle fuel produced at laboratory scale (AGR-1).
- Perform safety testing and PIE of both UCO and UO_2 TRISO particle fuel from prototypic production-scale equipment to obtain normal operation and potential accident condition performance data (AGR-2).
- Perform safety testing and PIE of representative UCO TRISO particle fuel containing DTF particles in support of fission product transport model development (AGR-3/4).
- Perform safety testing and PIE of the qualification test fuel to demonstrate that the reference fuel meets HTGR fuel performance requirements for normal operating conditions and potential accident conditions (AGR-5/6) and to obtain data needed for assessing the fuel performance margin to failure (AGR-7).

• Fuel Performance Modeling

- Improve the existing coated-particle material property database to support developing constitutive relations that describe the thermomechanical, thermophysical, and physiochemical behavior of coated particles.
- Develop a mechanistic fuel performance model for normal and off-normal HTGR conditions and benchmark against relevant performance data.

• Fission Product Transport and Source Term Determination

- Evaluate data from irradiation and safety testing of DTF fuel to characterize fission product release and transport from TRISO particle fuel into a fuel compact matrix and fuel element graphite under normal and off-normal HTGR conditions (AGR-3/4).

Understanding the relationship among the fuel fabrication process, fuel product properties, and in-reactor fuel performance is necessary. Fuel performance modeling is also addressed. The performance model is essential for several reasons, including guiding the future plant designer in establishing the core design and operating limits and in demonstrating to the licensing authority that the applicant has a thorough understanding of the in-service behavior of the fuel system and extrapolation of test results.

Irradiation and safety testing activities will also establish the operating margins for the fuel. For HTGR fuel, this means measuring the fuel performance at temperature, fast neutron exposure, and burnup levels at which the fuel begins to fail and release fission products in significant quantities, either during normal operation or under potential accident conditions. The AGR-7 experiment in irradiation test train AGR-5/6/7 will be designed so that some measurable level of fuel failure and/or fission product release is expected to occur.

Opportunities for collaboration will occur as others in the international community continue developing fuel for an HTGR. A Very high temperature reactor (VHTR) Fuel and Fuel Cycle Project Management Board has been established under the Generation IV International Forum (GIF) VHTR project to identify areas of possible collaboration, and some activities are under way. These are mentioned in sections of the document below.

1.2 Program Status

As of April 30, 2017, the AGR Fuel program had completed the following major tasks:

- Fuel Fabrication
  - Utilized publicly available German coating process information and German fuel and material property data for development of a fuel design and fuel specifications.
  - Used German coating process information in conjunction with coating process information from the United States. MHTGR and NPR programs to establish a reference set of coating process parameters for laboratory-scale equipment, and verified that these coating parameters yield properties in the prismatic HTGR particle design that are equivalent to the German coating properties.
  - Developed a German-like laboratory-scale overcoating process and a laboratory-scale compacting process.
  - Improved on previous United States UCO fuel kernel fabrication methods (forming, calcining, carbothermic reduction, and sintering) that resulted in better carbon dispersion, kernel microstructure, and surface topography.
  - Designed a prototypic production-scale furnace retort and gas distributor nozzle for chemical vapor deposition of the TRISO coating layers and developed parameters that increased the charge mass about thirty-fold relative to the laboratory-scale coater.
  - Reestablished basic QC capability for coated-particle fuel and developed new QC methods (as required) for enhanced characterization of kernels, coatings, and compacts.
  - Identified an alternate means of producing resinated graphite (matrix) powder by dry jet milling of co-mingled components, thus eliminating methanol as a part of matrix production and reducing preparation time from days to hours. Demonstrated resinated graphite powder as a subcontracted commodity.
  - Identified a pharmaceutical process for overcoating TRISO fuel particles using a resinated (thermosetting) graphite powder, substituting water for methanol as the wetting agent, eliminating the potential generation of a Resource Conservation and Recovery Act hazardous mixed waste. Eliminated particle upgrading, recycle, and reclamation process steps, and reduced cycle time from days to a couple hours.
  - Developed an automated, multi-cavity compacting system with a volumetric feed system that is readily scalable for production.
  - Developed the thermal treatment schedule for compacts and demonstrated a combined cycle furnace for compact carbonization and heat treatment that produces compacts with excellent structure and high matrix density for very good thermal conductivity.

- Produced and characterized initial reference fuel particles and selected variants for shakedown irradiation testing (AGR-1).
- Updated reference fuel with a prototypic production-scale coater for fuel performance testing for UCO and UO_2 kernels (AGR-2).
- Produced compacts containing driver and DTF fuel particles for fission product transport testing (AGR-3/4).
- Produced low-enriched UCO kernels, TRISO particles, and fuel compacts for the AGR-5/6/7 experiments.

- Fuels and Materials Irradiations
  - Completed irradiation of the AGR-1 experiment compacts for 620 EFPDs to a maximum burnup of 19.6% FIMA in the Advanced Test Reactor (ATR) at INL with no fuel particle failures. The AGR-1 experiment was the shakedown test for irradiation, safety testing, and PIE of the initial reference fuel and selected variants from laboratory-scale equipment. The fuel used in the AGR-1 experiment was 19.8% enriched.
  - Completed refurbishment of the dry transfer cubicle at ATR for sizing of the AGR test trains in preparation for shipment from ATR.
  - Completed transport of the AGR-1 test train from ATR to the Materials and Fuels Complex (MFC) at INL to begin PIE in March 2010.
  - Collected, analyzed, and qualified millions of data points generated during AGR-1, AGR-2, and AGR-3/4 fuel fabrication, irradiation, and PIE for future support of Nuclear Regulatory Commission (NRC) licensing activities for TRISO particle fuel.
  - Completed irradiation of the AGR-2 experiment compacts for 559 EFPDs in ATR to a maximum burnup of 13.15% FIMA containing prototypic production-scale UCO and UO_2 fuel with no apparent fuel particle failures. The fuel used in the AGR-2 experiment was 14.0% enriched.
  - Completed shipment of the AGR-2 test train to MFC at INL to begin PIE in July 2014.
  - Completed irradiation of the AGR-3/4 test train in ATR with DTF fuel particles in April 2014 after 369 EFPDs of irradiation to a maximum burnup of 15.27%. The fuel used in the AGR-3/4 experiment was the same as that used in AGR-1 with 19.8% enrichment.
  - Completed transport of the AGR-3/4 test train from ATR to MFC in two shipments in the spring of 2015. Because of the size of the test train, it had to be cut into two pieces to fit into the shipping cask.
  - Completed final design of AGR-5/6/7 in September 2015.
  - Conducted the final design review of the AGR-5/6/7 test train in September 2015, with all comments received during and after the review incorporated into the final design documents in October 2015.
  - Initiated AGR-5/6/7 test train component fabrication in October 2015.
  - Completed and issued AGR-1, AGR-2, and AGR-3/4 Nuclear Data Management and Analysis System irradiation data qualification reports.

- Received fuel compacts from BWXT for the AGR-5/6/7 test train enabling the assembly phase to begin.
- Completed and issued the AGR-1 and AGR-2 as-run final irradiation reports.[9,10]
- Completed and issued the *AGR-3/4 Irradiation Test Final As-Run Report* [11] in FY 2015.
- Completed and issued the AGR-1 and AGR-2 safety test predictions reports.[12,13]
- Completed and issued the *Uncertainty Quantification of Calculated Temperatures for AGR-3/4 Experiment*.[14]

- Safety Testing and PIE
  - Performed nondestructive and destructive PIE of the AGR-1 capsule components and as-irradiated compacts.
  - Completed PIE and safety testing of the AGR-1 irradiated fuel and components, including 19 safety tests through September 30, 2016, at temperatures of 1600°C, 1700°C, or 1800°C for approximately 300 hours each. In one of these tests, three compacts were simultaneously tested in a varying temperature profile (minimum temperature 850°C, maximum temperature 1690°C) simulating a temperature transient during a core-conduction cool-down event.
  - Isolated and studied local SiC degradation responsible for SiC failure and cesium release.
  - Completed AGR-1 compact cross section ceramography for evaluating TRISO layer post-irradiation morphology (cracks, tears, inter-layer bonding, etc.)
  - Completed AGR-1 loose-particle ceramography for evaluating kernel swelling and buffer densification. Prepared and issued the AGR-1 PIE final report,[15] summarizing the findings of the AGR PIE and safety testing efforts performed at INL and ORNL.
  - Completed advanced electron microscopy and micro-analysis of as-fabricated, irradiated, and post-irradiation safety-tested AGR-1 particles at INL. Study focused on fission product transport phenomena (e.g. fission product precipitate compositions and distributions in TRISO layers). Issued final report on this work.
  - Completed disassembly, metrology, and gamma scanning of the test train and capsules, and initiated destructive PIE and safety testing of AGR-2 irradiated compacts.
  - Completed PIE of five as-irradiated AGR-2 compacts at ORNL (four UCO compacts and one UO_2 compact).
  - Completed safety testing and post-safety test destructive PIE of four AGR-2 irradiated compacts (two UO_2 and two UCO) at 1600°C for approximately 300 hours each.
  - Completed 1800°C safety testing and post-test destructive PIE of one irradiated AGR-2 UCO compact.
  - Completed five shipments of AGR-2 compacts from INL to ORNL for PIE and safety testing.
  - Completed AGR-2 compact ceramography for evaluating TRISO layer post-irradiation morphology (cracks, tears, inter-layer bonding, etc.).
  - Completed shipment of irradiated AGR-2 loose particles from ORNL to INL for ceramography to determine kernel swelling and buffer densification.

- Established contracts to perform PIE and safety testing on South African (PBMR) UO_2 compacts from AGR-2 Capsule 4
- Began PIE and safety testing of the AGR-3/4 experiment in March 2015 at INL with test train disassembly, metrology, and gamma scanning of capsules and compacts.
- Issued AGR-3/4 first-look report detailing component metrology.
- Revised AGR-3/4 as-run thermal analysis based on results of component metrology.
- Developed equipment and methods for destructive PIE at INL of AGR-3/4 graphite rings containing fission products.
- Installed necessary equipment in radiation hot cell and initiated radial deconsolidation of irradiated AGR-3/4 compacts at INL.
- Initiated re-irradiation and safety testing of irradiated loose particles at INL. This enables study of I-131 release from AGR fuels.
- Completed conceptual design review of furnace and associated systems for safety testing irradiated fuels under air/moisture-ingress conditions at INL.
- Initiated bench-top testing of furnace and related systems to support development of the hot-cell air/moisture-ingress furnace being designed at INL.

- Fuel Performance Modeling
  - Developed the fuel performance modeling code, Particle Fuel Model (PARFUME), which has been used for irradiation pre-test and safety test predictions a
  - Completed and issued two comparison reports of fuel performance based on the experimental results of AGR-1 PIE[16] and AGR-1 safety tests with PARFUME pre-test predictions.[17]
- Fission Product Transport and Source Term
  - Completed hydrogen and tritium permeation measurements in the HTGR candidate high nickel superalloys Incoloy 800H, Inconel 617, and Haynes 230.
  - Performed gamma spectrometry measuremets of Ag-110m, Cs-134, and Cs-137 distributions in AGR-3/4 matrix and graphite rings

1.3 NRC Assessment Status

In 2014, the NRC staff completed its assessment of two previously submitted NGNP licensing white papers that described the AGR Fuel program and the approach to determining mechanistic source terms, an approach that relied extensively on data being obtained in the AGR Fuel program.[18,19] The results of the assessment were documented and transmitted to DOE via a letter with two enclosures.[20] The enclosures provided feedback on key licensing issues that are closely tied to the AGR Fuel program, its approach to fuel development and qualification, and to mechanistic source terms.

In its assessment, the NRC found:

> *In summary, the staff views the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms as generally reasonable. The staff observes that the fuel development and testing activities completed to date in the AGR Fuel Program appear to have been conducted in a rigorous manner and*

with early results that show promise towards demonstrating much of the desired retention capability of the TRISO particle fuel developed for NGNP. Moreover, the staff believes that the planned scope of activities in the AGR Fuel Program is reasonably complete within the context of pre-prototype fuel testing.

Regarding fission product transport phenomena and the collection of supporting data in the AGR Fuel program, the NRC found:

> *The NRC staff's FQ-MST [Fuel Qualification-Mechanistic Source Term] assessment report concludes, with caveats, that DOE/INL's ongoing and planned testing and research activities for NGNP fuel qualification and mechanistic source terms development appears to constitute a reasonable approach to establishing a technical basis for the identification and evaluation of key HTGR fission product transport phenomena and associated uncertainties. The staff expects more information on release and transport phenomena through event-sequence-specific pathways to be developed as DOE/INL's activities in these areas proceed.*

The "caveats" noted in the NRC assessment pertain primarily to the NRC staff's perceived need for fuel surveillance and testing of fuel fabricated in the production fuel facility and taken from the initial core of the prototype HTGR. Examples of more specific caveats are provided from the following staff finding:

> *The staff acknowledges that the AGR Fuel Program includes significant ongoing and planned research efforts to investigate the poorly understood phenomenology of silver and palladium interactions with TRISO coating layers. DOE/INL has stated that these research efforts may include examinations on fuel samples irradiated in the ATR at temperatures significantly above those normally expected during irradiation in an NGNP core. The staff would consider new insights emerging from such investigations in evaluating the potential fuel performance uncertainties associated with the initially unmet need for test data from real-time fuel irradiations in an HTGR neutron spectrum.*

Regarding plans to characterize the effects of air and moisture ingress on oxidation of fuel element graphite and matrix materials,[21] the NRC staff noted:

> *The staff finds that the submitted experiment plan presents a reasonable approach for developing the data needed to model how air and moisture ingress can affect NGNP TRISO fuel performance and fission product transport. Ensuring that the experiments adequately envelope all LBEs [licensing-basis events] that involve air or moisture ingress in the final NGNP design will be important.*

These significant NRC findings indicate that the AGR Fuel program is on track to meet its goal of providing a fuel qualification data set in support of the licensing and operation of an HTGR.

2. GOALS, ASSUMPTIONS, AND OBJECTIVES

This section presents the overall set of programmatic goals, assumptions, and objectives developed to guide the preparation of this plan. The scope of the technical program plan is divided into five program elements:

1. Fuel fabrication
2. Fuel and materials irradiation
3. Safety testing and PIE
4. Fuel performance modeling
5. Fission product transport and source term.

Detailed goals, assumptions, and objectives developed to guide the planning of each of these program elements are discussed in Section 3. A high level set of goals, assumptions, and objectives from the perspective of the overall AGR Fuel program are identified in Subsections 2.1, 2.2, and 2.3.

2.1 Overall Program Goals

The overall goals for the AGR Fuel program are to:

- Provide a fuel qualification data set to support licensing and operating a prismatic HTGR. HTGR fuel performance demonstration and qualification compose the longest-duration R&D task required for design and licensing. The fuel is to be demonstrated and qualified for service conditions encompassing expected normal operating and potential accident conditions.
- Support deployment of the HTGR for hydrogen and energy production in the United States by reducing the market entry risks posed by technical uncertainties associated with fuel production and qualification.
- Use international collaboration mechanisms to extend the value of DOE-NE resources (primarily through GIF VHTR-related activities).
- Support establishing a domestic TRISO particle fuel manufacturing capability for fabricating demonstration and qualification experiment fuel.

Fuel qualification is herein defined as demonstrating the robust performance and efficacy of the reference TRISO particle fuel by producing experimental data and analytical results.

2.2 Overall Program Assumptions

Overall program assumptions are as follows:

- Government and potential industry co-sponsors of the HTGR recognize that a stable, long-term, disciplined, fuel-development and qualification effort offers the greatest probability of success.
- Fission product retention in coated-particle fuel at the level demonstrated by the German program in the late 1980s (proof test composite EUO 2358-2365) meets the needs of the United States program.
- Proposed HTGR designs may impose more demanding service conditions than the German high-temperature reactor (HTR) Modül and require testing of a fuel based on the prismatic HTGR design and the German coating process.

- It is technically feasible to reestablish, at a reasonable cost, a production capability in the United States that is equivalent to the German capability.
- A base technology program aimed at reestablishing the capability to fabricate and test fuel, with a follow-on goal of improving the technology to the point where it can support economic deployment of an HTGR, is the lowest-risk approach to achieving the program goals.
- The target peak time-averaged fuel temperature (1250°C) can support HTGR operation at least to the lower end of the anticipated design core-outlet, helium-coolant temperature range (750 to 800°C).
- Annual DOE funding allocations that are less than those required to support the planned work scope included in the life-cycle baseline (LCB), as shown in Section 4 (Figures 4, 5, and 6), will impact plans presented here, causing delays to the schedule or reductions in planned work scope.
- Results of the AGR Fuel program will be responsive to the design data needs of the reactor and fuel vendors and to the NRC's fundamental licensing analysis data needs.
- Radiologically significant reactivity transients (those capable of compromising fuel integrity) are precluded by design; consequently, fuel performance and fission product release under these conditions need not be experimentally characterized.
- Activities relating to the licensing of a fuel vendor's product by the NRC Office of Nuclear Reactor Regulation and meeting the NRC mandate of 10 CFR 50, Appendix B[22].
- No major programmatic or technical difficulties that could impact the LCB or schedule will be encountered during the fuel development, irradiation testing, or PIE and safety testing. The LCB is the established baseline of all activities included in the AGR Fuel program, including schedule, performance duration, estimated costs, and logic ties.

2.3 Overall Program Objectives

Key objectives for the AGR Fuel program are delineated below.

- Establish an HTGR TRISO fuel development and qualification program that will:
  - Address the generic issues previously identified by NRC staff members in their pre-application reviews of the prismatic MHTGR and the pebble-bed modular reactor concepts.
  - Produce fuel fabrication specifications that meet the anticipated performance requirements of the reactor designer.
  - Prepare a fuel design data manual that captures the correlations and uncertainty estimates for fuel performance and fission product transport that are developed under the AGR Fuel program for this UCO TRISO particle fuel.
  - Support establishment of a domestic industrial/commercial capability to fabricate high-quality TRISO particle fuel using United States experience and nonproprietary German coating product characteristics and process data.
  - Improve understanding of the fabrication process, its impact on as-fabricated fuel properties, and their impacts on in-reactor performance.
  - Support establishing the domestic industrial/commercial capability to manufacture fuel elements consistent with HTGR design options.

- Produce or obtain TRISO particle fuel samples, as needed, to support required testing.
- Complete the design and fabrication of in-reactor test trains for irradiation testing of TRISO particle fuel.
- Develop and qualify TRISO particle fuel by generating and presenting statistically sufficient irradiation and PIE data under normal operating conditions, and develop and qualify safety testing data under potential accident conditions consistent with anticipated design requirements.
- Demonstrate a sufficient margin to failure for this fuel form under normal operating and potential accident conditions.
- Enhance understanding of fuel behavior and fission product transport to improve the fuel performance and fission product transport models, so that fuel behavior and fission product transport under normal operating, operational transient, and potential accident conditions can be predicted to accuracies within quantified uncertainty limits.
- Develop pertinent fuel process information that can be used by HTGR fuel vendors to select and implement fuel fabrication processes.
- Develop pertinent fuel qualification information that can be used by HTGR fuel vendors in topical reports supporting HTGR licensing.
- Focus fuel fabrication process development on low-enriched UCO TRISO particle fuel.
- Implement this plan in accordance with the DOE QA requirements specified in 10 CFR 830, "Nuclear Safety Management," Subpart A, Quality Assurance Requirements, [23] and in DOE O 414.1D, "Quality Assurance."[24]

All activities that have direct input to the irradiation test specimen fabrication, irradiation campaigns, and safety testing will be conducted in accordance with national consensus standard Nuclear Quality Assurance (NQA)-1-2008, Addenda 1a 2009, "Quality Assurance Requirements for Nuclear Facility Applications,"[25] published by the American Society of Mechanical Engineers. Each participating organization shall prepare specific QA plans for its assigned scope of work and may prepare additional project-specific plans for individual work breakdown structure (WBS) elements as appropriate.

3. PROGRAM ELEMENTS

This section summarizes detailed goals, assumptions, and objectives associated with the individual program elements and the activities performed or required to meet these and the high-level goals and objectives identified in Section 0. Program elements discussed in more detail below include fuel fabrication, fuel and materials irradiation, PIE and safety testing, fuel performance modeling, and fission product transport and source term.

3.1 Fuel Fabrication

3.1.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives specific to this program element are as follows.

3.1.1.1 Goals

1. Establish a production-scale TRISO particle fuel fabrication technology in the United States that is capable of producing fuel at a quality level at least equivalent to those of German fuel particles from

composite EUO 2358-2365.

2. Develop a fundamental understanding of the relationships among fuel fabrication process parameters, fuel product properties, and fuel performance under normal operating and potential accident conditions.
3. Develop appropriately automated fuel fabrication technology suitable for mass production of coated-particle fuel at an acceptable cost and at acceptable levels of high quality and consistency.
4. Establish fuel process and product specifications that define the requirements that the as-fabricated fuel must meet to ensure acceptable performance within the selected envelope of HTGR service conditions.
5. Develop and document the manufacturing processes required to meet the fuel process and product specifications that will be developed to satisfy Goals 2 and 3 above.

3.1.1.2 Assumptions

1. The coated-particle design to be qualified in the AGR Fuel program will be based on the most stringent performance requirements for two different types of HTGRs (pebble-bed and prismatic). This approach will result in the qualification of a fuel performance envelope that can be used by either HTGR technology.
2. Fuel capable of acceptable normal operation and potential accident condition performance up to a target peak time-averaged fuel temperature of 1250°C in normal operation can support HTGR operation within a substantial portion of the anticipated core-outlet helium-coolant temperature range (750 to 800°C).
3. The capability to mass produce high-quality, coated-particle fuel elements economically is a prerequisite for commercial viability of HTGRs.
4. The low-enriched UO_2 particles qualified by the Germans in pebble-bed reactors for burnup to about 10% FIMA are not adequate for higher fuel burnup (16 to 22% FIMA), higher operating temperatures, and temperature gradient service in prismatic HTGRs.
5. Fuel particles made with low-enriched UCO kernels and having coating properties equivalent to those of German fuel particles from composite EUO 2358-2365 (that were irradiated in the HTR-Modül proof tests [High Flux Reactors K5 and K6 in Petten, Netherlands]) with no in-pile failures will perform well in fuel compacts under prismatic HTGR irradiation conditions.
6. The lowest-risk path to successful manufacturing of coated fuel particles is to closely replicate the proven German coating technology to the extent possible in a coated fuel particle design that incorporates the lessons learned from United States fabrication and irradiation experience to improve the coating process.

3.1.1.3 Objectives

1. Support establishment of and demonstrate coated-particle fuel fabrication capability from kernel production through fuel compact production.

2. Conduct fuel kernel process studies to optimize the UCO kernel fabrication process (carbon dispersion, broth chemistry, calcination, carburization, and sintering).
3. Conduct fuel-coating process studies to determine the capability to replicate the properties of German coated-particle fuel for HTGR fuel and to establish coating conditions that yield coating layers having microstructural properties and features comparable to the coating layers in the German fuel particles in proof-test composite EUO 2358-2365.
4. Develop a process suitable for large-scale fuel production that produces coating properties consistent with acceptable fuel performance. This will be accomplished using a coater that provides a coating environment similar to the German production-scale coater and has appropriate features for a production-scale coater (for loading, unloading, sampling material from the coater, and cleaning).
5. Develop additional QC methods to improve fuel characterization capabilities and results.
6. Fabricate fuel as needed for irradiation testing, including DTF fuel for fission product transport tests. The fuel shall meet the product requirements specified in the test fuel product specifications. These fuel product specifications will be based on specific objectives for each irradiation experiment.
7. Prepare a fuel product specification and process specification for large-scale HTGR fuel fabrication that defines all requirements the fuel must satisfy to ensure acceptable performance under HTGR normal operating and potential accident conditions.
8. Develop automation technologies that can be applied to fuel fabrication processes to the maximum extent practicable.

3.1.2 Scope of Fuel Fabrication

The ultimate fabrication goal for HTGR fuel is the economical production of high-quality kernels, TRISO-coated fuel particles, and compacts or pebbles that meet the fuel product specifications. The fuel fabrication activities described herein are intended to develop and qualify a fuel fabrication process that is the foundation for fabrication of production-scale, coated-particle fuel for HTGRs. These activities must optimize the process to achieve the required kernel, coated fuel particle, compact or pebble characteristics and quality. They must also result in scale-up of kernel production, coating, and compact or pebble fabrication processes.

Coated-particle fuel fabrication differs from light-water reactor fuel manufacturing. The fabrication process developed within the AGR Fuel program begins with low-enriched UCO kernels formed by the internal gelation process in which droplets of uranium-containing chemical broth are formed into gel spheres in a fluid medium. The resulting gel spheres are dried and sintered into hard ceramic particles yielding kernels of a controlled, consistent size and chemistry.

Fuel kernels are coated using a fluidized-bed chemical vapor deposition process. The coatings include a low-density carbon (buffer) layer, a high-density inner pyrolytic carbon (IPyC) layer, a SiC layer, and a high-density outer pyrolytic carbon (OPyC) layer. These coatings are designed to work together to make each fuel particle a mini pressure vessel that will maintain its integrity and retain fission products during normal reactor operation and potential accident conditions. The finished coated particle is a small (≤~1 mm outside diameter [OD]) carbon and ceramic sphere that is stable to temperatures well beyond 1600°C.

The coated fuel particles are formed and pressed into physical shapes for use in the reactor, as shown in Figure 1. For the prismatic reactor design, fuel particles are pressed into cylindrical-shaped compacts for insertion into large hexagonal graphite blocks, which are stacked in columns to form the reactor core. Fuel particles for the pebble-bed reactor design are pressed into tennis-ball-sized pebbles that may be recirculated in the reactor. For both designs, the particles are overcoated with a carbonaceous matrix composed of graphite powder and a resin binder, formed into the desired shape, carbonized, and treated at high temperature to provide a thermally stable material.

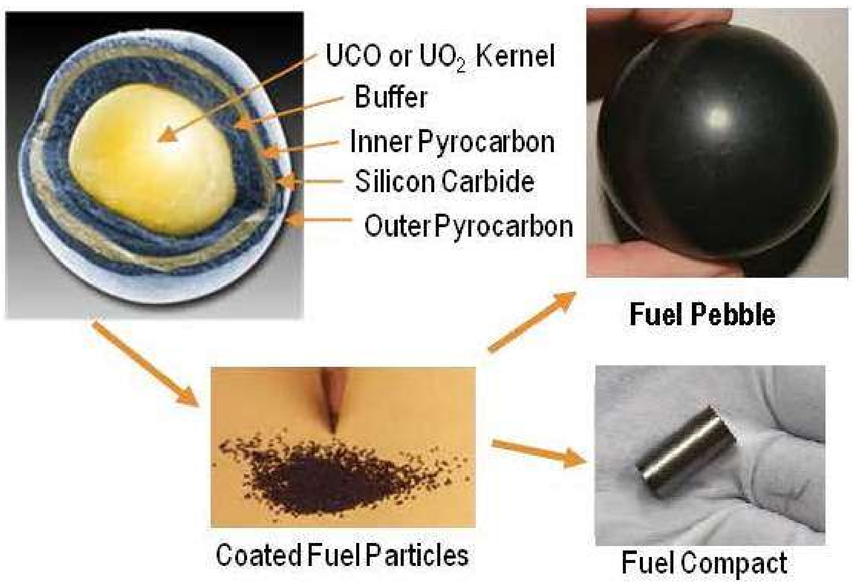

Figure 1. Formation of potential fuel forms.

The target quality level for coated-particle fuel is based on the quality level achieved in the German program in the late 1980s, with the EUO 2358–2365 fuel particle composite used in the HTR-Modül proof tests taken as a standard for comparison,[26] in combination with core-design-driven quality specifications derived during the GT-MHR conceptual design.[8] The AGR fuel fabrication effort was designed to expand the understanding of the relationship among kernel and coating properties, fabrication process conditions, and the irradiation performance of the fuel. The earlier United States and German manufacturing efforts and subsequent work in other national programs achieved a substantial level of understanding of these relationships, but additional work is required.

Fuel failures in United States MHTGR and NPR program irradiation tests have been analyzed [6,7] along with United States and German fuel fabrication processes and irradiation performance.[27] These studies suggest key differences between German and historical United States coating processes, and coating properties contribute to better irradiation performance. The most significant differences in the German processes are (1) a greater deposition rate of pyrocarbon layers, resulting in more isotropic coatings having greater stability to high fast neutron fluence under irradiation; (2) more intimate bonding of the IPyC and SiC coating layers; (3) continuous coating of all layers, resulting in less potential for as-manufactured defects and possible beneficial effects on coating properties; and (4) lower SiC coating temperature, resulting in smaller grain size. In addition, the German compacting process began with overcoating the coated particles with a graphite/resin blend to prevent particle-to-particle contact during pressing, versus a pitch-injection process used by earlier United States fuel programs. Thus, the starting point for fuel fabrication development was the United States kernel and compacting experience coupled with knowledge of the German coating and overcoating processes as supplemented by lessons learned from fuel technology development within the United States.

The work to produce TRISO particle fuel that meets the specifications has included kernel process development, coating process development, overcoating and compacting process development, advanced characterization and QC methods development, and process documentation. The scope of fuel manufacturing activities is summarized below.

3.1.2.1 Prepare Irradiation Test Fuel Specifications. Developing a fuel fabrication process and fabricating irradiation experiment fuel in a manner that complies with the QA requirements of NQA-1[25] is based on the specification of kernel, coated-particle, and compact properties and on key process parameters. Detailed product specifications, along with a limited set of process specifications affecting microstructure characteristics, which are known to be important to irradiation performance but cannot be fully characterized, are required for each irradiation experiment conducted within the program. These specifications include the parameters identified in Table 1. (Property specifications include properties for individual batches as well as for composited lots formed from multiple batches.)

Execution of this plan produced specifications for the fuel to be used in the series of AGR irradiation experiments, leading to a specification for fuel to be produced for an HTGR.

Table 1. Fuel specification parameters.

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Kernel Composite			
^{235}U Enrichment	X		
C/U ratio	X		
O/U ratio	X		
(C+O)/U ratio	X		
Individual Impurities (Li, Na, Ca, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Al, and Cl)	X		
Process Impurities (P, S)	X		
Envelope Density	X		
Diameter	X	X	X
Aspect Ratio		X	X
Microstructure	Visual Standard		
Coated-Particle Composite			
Buffer Density[b]	X		
IPyC Density[b]	X	X	X
Thickness (Buffer, IPyC, SiC, OPyC)	X	X	X
Density (SiC, OPyC)	X	X	X
Anisotropy (IPyC, OPyC)	X	X	X
Exposed Kernel Fraction	Measurement Only		
SiC Aspect Ratio		X	X
Defective IPyC Fraction	X		

Table 1. (continued).

Parameter	Mean	Critical Region[a]	Fraction in Critical Region
Defective SiC Fraction	Measurement Only		
Defective OPyC	X		
Pre-Burn Exposed Uranium	Measurement Only		
Post-Burn Exposed Uranium	Measurement Only		
SiC Soot Inclusions[c]	Measurement Only		
SiC Microstructure	Visual Standard		
Heat-Treated Compacts			
Uranium Loading	X		
Diameter		X	X
Length		X	X
Matrix Density	X		
Impurity Content (Fe, Cr, Mn, Co, Ni)	X	X	X
Impurity Content (Ca, Al, Ti, V)	X		
Heavy Metal Contamination Fraction			X
Exposed Kernel Fraction			X
Dispersed Uranium Fraction	X		
Defective SiC Fraction	X		
Defective OPyC Fraction	X		

a. The specification of a critical region boundary and the fraction of particles within the critical region are provided to limit the distribution tail of a property or, in the case of attribute properties, the subpopulation of abnormal particles with a specific defect.
b. Calculated from pooled characterization data for particle batches.
c. An indication of defects within the SiC layer, historically identified by General Atomics as "Gold Spots," but not detectible as such in the more opaque, finer-grained SiC shells.

3.1.2.2 Fuel Kernel Manufacturing. As discussed in Section 1, the AGR Fuel program elected to develop coated-particle fuel using a low-enriched UCO kernel to support the NGNP project. Low-enriched UCO kernels had been produced for earlier irradiation testing in the United States by General Atomics (GA)[28,29] and BWX Technologies Inc. (BWXT).[30] Currently, BWXT possesses the only commercial domestic fuel fabrication facilities that can handle uranium enrichment levels in excess of 5%. The BWXT internal gelation low-enriched UCO kernel production process was selected for the AGR Fuel program with the understanding that additional process development would be needed to improve the overall quality of the product and adjust for the kernel diameters specified. Thus, the scope of fuel kernel manufacturing included the following elements.

- Process development:
  - Achieve specified kernel density
  - Improve carbon dispersion in the acid-deficient uranyl nitrate solution used in kernel formation
  - Optimize the sintering process

- Reduce process variability

- Produce natural UCO kernels to support coating process development
- Produce low-enriched UCO kernels for use in production of fuels to be irradiated.

3.1.2.3 Coating Process Development. When the AGR Fuel program began, no active coating process facilities existed within the United States. The GA coater used to coat the fuel irradiated in High Flux Isotope Reactor (HFIR) Removable Beryllium-21 (HRB-21) and NPR capsules had been shut down for more than a decade, and the facility had been completely dismantled. Small coaters remained at ORNL and BWXT, but neither had been operational for the production of TRISO fuel for many years. Thus, United States TRISO particle coating capability needed to be reestablished. In addition, root cause assessments of the HRB-21 and NPR capsule fuel particle failures[6,7,27] indicated a need to adjust the coating process parameters to change the IPyC, SiC, and OPyC layer microstructures. Given the successful performance of pyrolytic carbon (PyC) and SiC coatings produced by the German program, a primary objective was to identify process parameters that would produce coating characteristics equivalent to German coatings. Another objective was developing an improved understanding of the relationship between coating process parameters and key coating characteristics known to be important for irradiation performance.

A relatively large number of coating runs was required during initial coating development to obtain process conditions and durations that produced the desired coating properties and thicknesses. These runs were conducted in a laboratory-scale coater to limit the cost and quantity of materials required, as well as to minimize wastes. The assessments noted above concluded that a focus on limiting uranium dispersion during application of the SiC layer by reducing permeability of the IPyC layer resulted in an IPyC layer that was prone to failure during irradiation. Thus, the process development scope included a study of the relationship among IPyC coating conditions, IPyC layer permeability, and IPyC properties that influenced irradiation performance (density, anisotropy, and surface-connected porosity).

Laboratory-scale coater runs established the process conditions needed to produce particles that met the specifications and improved understanding of the relationship between process parameters and key coating properties, but some uncertainties regarding the relationship between properties and irradiation performance remained. Therefore, a reference fuel specification plus variations in key coating parameters were needed to provide confidence in achieving acceptable performance in the first irradiation experiment (AGR-1). Use of multiple coated-particle types (baseline and three variants) while meeting the AGR Fuel program schedule and funding constraints required that the fuel for the first irradiation (AGR-1) and initial fission product transport irradiations (AGR-3/4) be produced in the ORNL laboratory-scale coater.

Producing the quantities of fuel required to support initial HTGR operation (first core) will require a larger coater, so fuel qualification based on fuel produced in a larger coater was a goal. Thus, coater scale-up issues needed to be addressed in the context of defining the coater size and configuration for producing the particles used in subsequent irradiation tests and producing fuel for the initial HTGR. A 6-in.-diameter coater was selected as the “large” size for the coating scale-up effort. Although a larger coater (or multiple 6-in. coaters) would likely be needed for large-scale commercial fuel manufacturing to support deployment of multiple HTGR plants, a 6-in.-diameter coater was selected as a reasonable size for the initial scale-up effort and adequate for production of fuel for a first-of-a-kind HTGR. The results of the small coater operation were used to reduce the number of large coater runs needed to achieve the specified coating properties, but process development scope was needed to define the large coater process conditions. The large coater was then used to produce the coated particles needed for the AGR-2 irradiation experiment and is being used to produce the fuel needed for the AGR-5/6/7 irradiation experiments.

3.1.2.4 Compacting Process Development. Historically, the United States compacting technology has used a thermoplastic matrix consisting of petroleum pitch mixed with graphite powder and injected into a mold containing fuel particles to make compacts. The injection process can result in high stresses on the particles where point-to-point contact occurs, which is a potential mechanism for particle failure. The compacts were also packed in alumina powder during carbonization to prevent them from losing their shape. The raw materials used in the thermoplastic matrix had relatively high concentrations of metallic impurities that were highly reactive with SiC at high temperatures. The alumina powder used in the carbonization process was another source of impurities that potentially attacked the SiC layer.

Shortcomings in the historical United States compacting process were addressed during AGR-1 laboratory-scale compact development by using purified graphite and resin material and a German-like overcoating process to prevent particle-to-particle contact during pressing. The selected thermoplastic resin was similar to one of the resins used successfully by the German program and eliminated the need for compact support during carbonization.

A thermosetting resin-based matrix process was selected for production-scale fuel manufacturing. This thermosetting resin-based matrix was also formulated from raw materials having low levels of impurities, and it yields stronger, less friable compacts. The thermosetting matrix process can also involve lower compacting forces, thereby reducing the potential for damage while allowing for increased matrix density.

Compacting process development scope included:

- Replicating the matrix formulation of a German thermosetting resin/graphite blend
- Jet-milling the resin, graphite, and hexamethylenetetramine mixture to provide a very uniform matrix supply without the use of methanol to solvate the resin
- Substituting water for methanol during TRISO particle overcoating aimed at eliminating the generation of a Resource Conservation and Recovery Act mixed hazardous waste, and necessary development of a waste disposition path
- Establishing prototypic production-scale overcoating equipment and process conditions needed to uniformly overcoat particles with the matrix
- Establishing the automated pressing equipment and process conditions needed to form the overcoated particles into compacts, and performing carbonization and final heat treatment in a furnace capable of combining the two steps
- Producing compacts needed for characterization and irradiation in the AGR Fuel program's final AGR-5/6/7 irradiation test.

3.1.2.5 QC Methods Development and Application. QC methods were needed to demonstrate that the fuel fabricated for the AGR Fuel program complied with the product specifications. As with the coating process, facilities were unavailable to measure the properties identified in Engineering Design File (EDF)-4380, "AGR-1 Fuel Product Specification and Characterization Guidance,"[31] at the required confidence levels (typically 95% confidence). Therefore, the development of QC methods involved reestablishing traditional characterization procedures at ORNL and BWXT and developing advanced QC methods, mainly at ORNL. The following QC capabilities were needed for the inspection and testing of kernels, TRISO particle fuel, and compacts to demonstrate compliance with fuel product specifications:

- Chemistry of kernel batches and composites (carbon, oxygen, uranium, and 15 impurities)
- Kernel ^{235}U enrichment
- Ceramography to provide images for coated particle analysis
- Hardware and software upgrades to the ORNL ellipsometer (2-MGEM)
- Automated image analysis for kernel and particle diameter, aspect ratio, and coating thickness measurements
- Density gradient columns for PyC and SiC sink-float density measurements
- Mercury porosimetry for measuring kernel and buffer envelope density and for PyC surface-connected porosity measurements
- An improved technique for measuring PyC coating anisotropy
- Compact measurements, including length, diameter, mean uranium loading, total mass, matrix density, and defective IPyC and OPyC coating fractions
- Leach-burn-leach (LBL) testing of fuel compacts to determine the exposed kernel, dispersed uranium, and defective SiC fractions, and the quantity of specified impurities outside the SiC layer
- X-ray analysis for detecting uranium dispersion in coated particles
- Inspection of particles for soot inclusions and other abnormalities in the SiC layer
- X-ray analysis for detecting gross soot inclusions and misshapen particles in addition to defective IPyC
- X-ray tomography for improved characterization of the internal structure of unirradiated and irradiated fuel particles.

3.1.2.6 Fuel Product and Process Documentation. The description of fuel fabrication development, irradiation, PIE, and safety testing in this plan, when combined with additional reactor design information, provides the information to finalize the top-tier fuel product specifications that define requirements for fuel to be used in an HTGR. Additional reports will be produced to document process and QC development as well as pre- and post-irradiation data for all irradiation tests. The Reports compiling process-development and product data compilation reports will provide a basis for the final process parameters necessary to fabricate fuel that consistently meets the fuel product specifications and performance requirements of an HTGR, and the allowable process variations (to the extent determined by the process development tasks).

3.2 Fuel and Materials Irradiation

Irradiation testing of coated-particle fuels occurred routinely in the United States from the 1960s through the early 1990s. Materials test reactors are still in operation, and personnel experienced with all aspects of irradiation test train design, assembly, and monitoring are active at INL and ORNL. ORNL irradiated fuel for the MHTGR,[32] and both laboratories were involved in irradiation testing of NPR and MHTGR fuel in the early 1990s. ATR at INL and HFIR at ORNL are capable of irradiation testing of AGR fuels. ATR was selected in large part because of the availability of an irradiation location that has a very close match to the nominal gas reactor conditions, resulting in an excellent approximation of HTGR burnup and fast fluence.

3.2.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives of fuel and materials irradiation are as follows.

3.2.1.1 Goals

- Provide data for fuel performance during irradiation to support fuel process development, qualify fuel for normal operating conditions, and support development and validation of fuel performance and fission product transport models and codes.
- Provide irradiated fuel and materials for PIE and safety testing.

3.2.1.2 Assumptions

- Accelerated irradiation in ATR (up to a maximum of three times real-time in terms of both power and fast flux) is equivalent to or is conservative relative to real-time irradiation.
- Developmental fuel fabrication capability is established to provide fuel samples for near-term irradiation.
- Limited material sample irradiations can be conducted in conjunction with fuel irradiation without requiring additional test trains.
- Radiologically significant reactivity transients are precluded by inherent characteristics of the design, so no reactivity insertion accident testing is planned.
- Fuel fabrication capability is established to provide fuel samples representative of high-volume production for qualification testing.
- Waste activated/contaminated metal (leadout, gas lines, thermocouple [TC] leads, etc.) will be staged in the ATR canal until a cleanup campaign is conducted by ATR Operations. There is no additional cost to the AGR Fuel program for disposal of this waste.

3.2.1.3 Objectives

- Establish the range of irradiation conditions (power, burnup, flux, fluence, temperature, and environment) based on the needs of the reactor designs and the needs of the associated topical report licensing strategy to qualify fuel for normal operation.
- Establish allowed tolerances on control of irradiation conditions.
- Complete design and fabrication of test trains for irradiation testing of TRISO particle fuel.
- Establish and conduct a fuel and materials-irradiation activity that will provide:
  - Independently controlled and monitored capsules within an irradiation test train.
  - Control capability to maintain conditions within the planned tolerances.
  - Online monitoring of release of indicator fission product gases such as krypton and xenon isotopes.
  - A test train design that will allow post-irradiation measurement of metallic fission product release, such as silver (Ag), Cs, and strontium (Sr), from fuel in each capsule during irradiation.

- Sufficient data to qualify the fuel for normal operation over the required range of irradiation conditions and to support code and model development and validation.
- Irradiated fuel and material specimens required to support PIE, post-irradiation phenomenological testing, and safety testing activities.

3.2.2 Scope of Fuel and Materials Irradiation

In producing the original version of the technical program plan, the fuel irradiation working group developed a description of the tasks associated with irradiation testing of a representative test train in ATR. Even though the details of test train internals, test articles, and control parameters will vary depending on the requirements for a given irradiation, as defined in the applicable experiment specification, the basic tasks remain the same. This task list, along with corresponding deliverables and interfaces with other activities, has served as the basis for schedule and cost estimates for irradiation testing. The following tasks were identified:

1. *Experiment specification*. This task will specify the test articles, irradiation conditions, and results needed to support fuel fabrication, model development, and plant design and licensing. The experiment specification document will include a definition of test articles to be included in the test train, required operating conditions (including tolerances), and required data (including accuracies) to be produced by the experiment.
2. *Test train and supporting systems' technical and functional requirements*. This task will establish the detailed requirements necessary to proceed with test train and supporting systems' design in accordance with the experiment specification. The resulting document will include general design requirements associated with the service conditions of the test train in the reactor, design and functional requirements specific to the test train and its supporting systems, and provisions for QA. The document will also include the requirements placed on the experiment by ATR necessary to meet ATR technical specifications and safety analysis report requirements (materials allowed, departure from nucleate boiling ratio, flow instability ratio, etc.).
3. *Test train and supporting systems' design*. This task will establish the detailed design and procurement specifications necessary to proceed with test train fabrication/assembly and establish the needed supporting systems for either a new test train design or replication of a proven test train design.
4. *Test train and supporting systems' fabrication/assembly*. This task includes procuring or fabricating test train components in accordance with the specifications; installation of the components; refurbishment of supporting systems, as necessary; and assembly of the test train, including the test articles, so that it is ready for insertion into the reactor.
5. *Approval of test articles*. This task includes the receipt, inspection, and QA acceptance of all test articles (compacts, pebbles, loose particles, and/or material samples) to be incorporated into the test train.
6. *Review/approval of final design and fabrication data packages*. This task includes review and concurrence by affected program participants.
7. *Irradiation*. This task addresses all activities associated with irradiation of the test train, including preparation of detailed operating procedures for test train handling during insertion and removal,

preparation of experiment safety analysis documentation, preparation of the experiment safety assurance package, test train insertion into and removal from the reactor, operation of the fission product monitoring system, technical support, operation of data-acquisition systems, documentation of conditions and results of irradiation (including a near-real-time remote data-acquisition capability), and placement of the test train in the ATR canal for cooldown once irradiation is completed.

8. *Cooldown and preparation for shipping*. This task addresses storage (estimated to be about 90 days) of the test train in the ATR canal until the decay heat and radiation levels (from fuel and activated metal) are sufficiently low to proceed with sizing of the test train in the dry transfer cubicle at ATR. Preparations include development of mockups of the test train, development of detailed operating procedures for the sizing activity, dry runs of the planned sizing evolution, actual sizing of the test train, and loading of the shipping cask or package for shipment of the test train to MFC for PIE and safety testing. A GE-2000 cask was leased for shipment of the AGR-1, AGR-2, and AGR-3/4 test trains. The AGR-5/6/7 test train may be shipped in an alternative shipping package. The size of the test train will determine the shipping configuration and number of shipments required.

9. *Waste disposition*. The leadout and test train cuttings are waste forms associated with the AGR irradiations. The lower non-fueled section of the test train is cut off in the ATR canal and temporarily disposed of there as waste. The upper non-fueled section of the test train is cut off in the dry transfer cubicle and then removed and placed in the ATR canal as waste. The cuttings from the test train sizing evolutions are captured in a tray and placed in the ATR canal as waste. These waste sections are dispositioned with other activated/contaminated metal during the course of routine cleanup activities of the ATR canal. The test train gas lines and TC leads are left in the leadout and dispositioned at the same time.

3.2.3 AGR Irradiations

The number and type of test trains to be irradiated were planned based on the needs of the fuel manufacturing, fuel performance modeling, and fission product transport activities. The selected test train concept used in the first two irradiations, AGR-1 and AGR-2, were placed in large B positions of ATR. The AGR-1 "shakedown" test train contained six capsules independently controlled for temperature and separately monitored for fission product gas release, with each capsule containing twelve 1-in.-long by ½-in.-diameter compacts. The AGR-2 test train contained six capsules independently controlled for temperature and separately monitored for fission product gas release. United States made UCO fuel was included in three capsules, and UO_2 fuel was included in one capsule. The fifth capsule contained French UO_2 fuel, and the sixth capsule contained South African UO_2 fuel. To increase the capacity for irradiation of fuel and decrease the duration of its irradiation, the AGR-3/4 test train was designed for the ATR northeast flux trap (NEFT) position. The AGR-3/4 test train contained 12 capsules, with each capsule containing four ½-in.-long by ½-in.-diameter compacts. The AGR-3/4 test train capsules were independently controlled for temperature and separately monitored for fission product gas release. The design and configuration for the AGR-5/6/7 experiment will consist of five capsules of varying lengths containing 1-in.-long by ½-in.-diameter compacts with irradiation planned in the NEFT.

The B positions in ATR are located in four triangular arrays, with each array comprising two small B positions and one large B position. The arrow labeled "Small B Position" in Figure 2 points to one of the eight small B positions, which are adjacent to the driver fuel. The arrow labeled "Large B Position" points to the large B position in the east quadrant of the core. This B position is one of four located farther from the driver fuel in the beryllium reflector that have a higher ratio of thermal-to-fast flux. Reactor physics calculations conducted by INL for the large B positions show a ratio of burnup to fast fluence that is well matched to expected HTGR conditions. The physics calculations are refined when the actual fuel loadings are known for each test train. The AGR-1 and AGR-2 experiment irradiations were scheduled for about 3 years depending on the ATR operating schedule to reach the planned approximately 600 EFPDs of irradiation. In actuality, the AGR-1 experiment was irradiated for 620 EFPDs starting in December 2006 and ending in November 2009, and the AGR-2 experiment was irradiated for 559 EFPDs starting in June 2010 and ending in October 2013. The AGR-3/4 experiments completed irradiation in the NEFT in April 2014 after 369 EFPDs, having initiated irradiation in December 2011 and reaching the target burnup levels for all capsules. The NEFT irradiation position used for the AGR-3/4 test train (arrow at top upper right side of Figure 2) can accommodate larger test trains at increased power levels to reduce irradiation times. The NEFT is also planned to be used for the AGR-5/6/7 experiments. Preliminary calculations indicate irradiation times on the order of about 3 years depending on the ATR operating schedule in the NEFT location to reach the planned 500 to 550 EFPDs of irradiation needed to achieve targeted burnup for the AGR-5/6/7 experiments.

Continuous gas monitoring capability for the AGR-1 and AGR-2 experiment capsules within the test train was provided by a set of six dedicated fission product monitors plus one online operating spare. For the AGR-3/4 experiments irradiated in the NEFT, continuous gas monitoring was provided by 12 dedicated fission product monitors plus two online operating spares. The AGR-5/6/7 experiments are designed for five capsules, and continuous gas monitoring will be provided by five dedicated fission product monitors plus two online operating spares.

The seven experiments were identified based on discussions among the working groups during the course of developing the original plan. Program budget constraints and further development of the test train designs have altered the type of test trains that were initially planned to be used for individual irradiations. For example, it was decided to conduct the AGR-3/4 and AGR-5/6/7 irradiation testing in the NEFT within ATR; the NEFT accommodates a larger test train and has a higher acceleration factor to shorten the irradiation schedule timeframe.

3.2.3.1 Shakedown/Early Fuel Experiment (AGR-1). This multi-monitored capsule test train included six capsules, each containing 12 compacts made from TRISO particles produced in a small laboratory-scale (2-in.) coater in conjunction with fuel process development. This irradiation experiment provided experience with a multi-monitored test train design, fabrication, and operation, which facilitated the design, fabrication, and operation of subsequent irradiation experiments. Having been successfully taken to estimated design burnup and fast fluence, AGR-1 has provided data on irradiated fuel performance for baseline and fuel variants selected based on data from fuel process development and existing irradiation experience. The early data on performance of fuel variants supported the selection of a reference fuel for the AGR-2 irradiation experiment and development of an improved fundamental understanding of the relationship among the fuel fabrication process, as-fabricated fuel properties, normal operation, and potential accident condition performance.

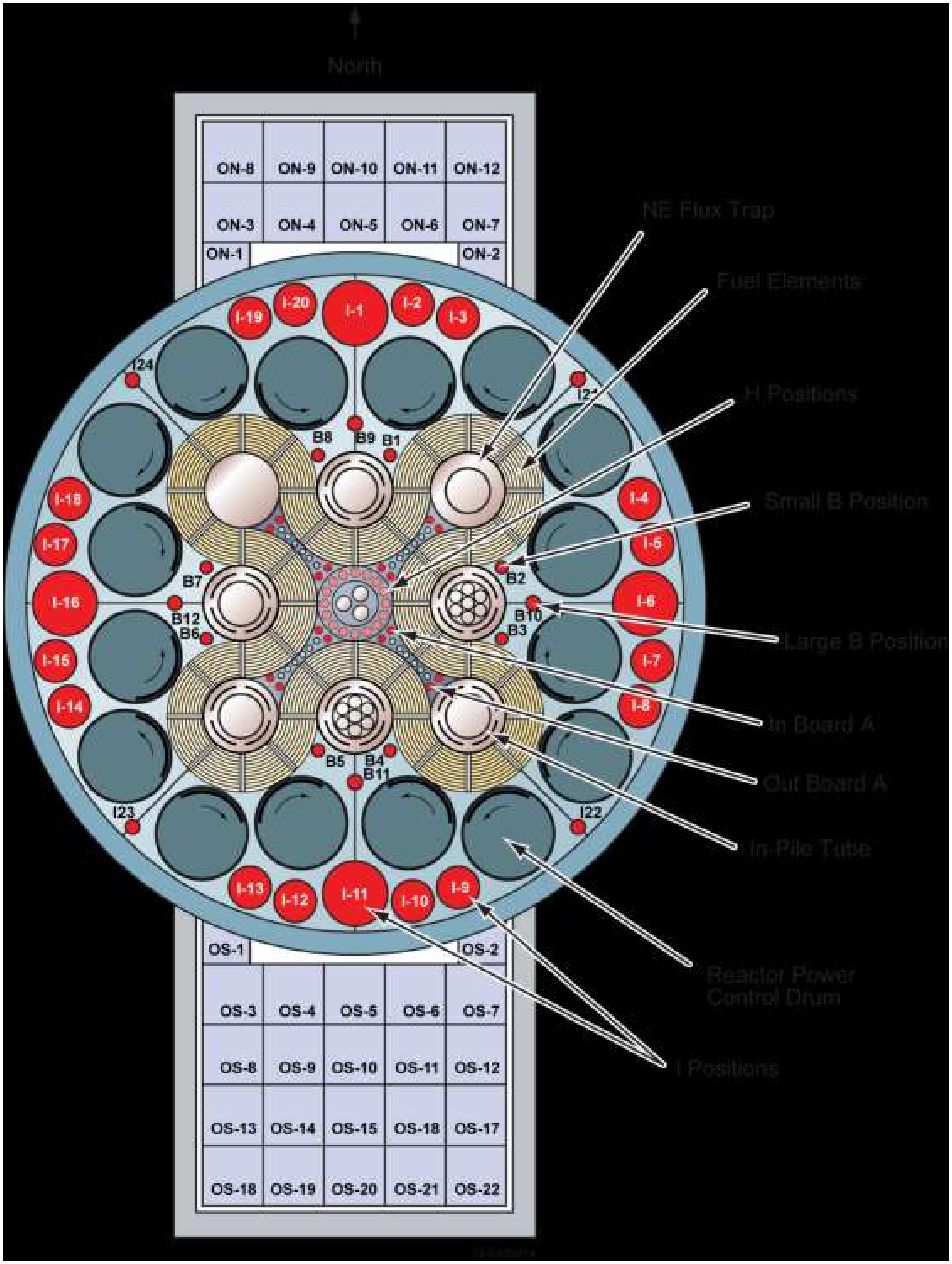

Figure 2. ATR cross section.

3.2.3.2 Performance Test Fuel Experiment (AGR-2). This multi-monitored capsule test train included three capsules of 12 compacts (fabricated at laboratory-scale using the same process conditions as AGR-1), each containing United States. UCO particles made in a production-scale 6-in. coater using process conditions derived from the production of AGR-1 Variant 3 (SiC layer produced using a mixture of hydrogen and argon diluent gases). The UCO compacts were subjected to a range of burnups and temperatures exceeding anticipated reactor service conditions in all three capsules. The test train also included three additional capsules of six to 12 compacts, each containing UO_2 particles produced independently by three program participants (BWXT, Westinghouse/Pebble Bed Modular Reactor SOC Ltd., and Commissariat a l'Energie Atomique/AREVA), with UO_2 particles from BWXT and Pebble Bed Modular Reactor SOC Ltd. also compacted using the AGR-1 laboratory-scale process. The range of burnups and temperatures in these capsules exceeded anticipated pebble-bed reactor service conditions. This test train provided irradiated fuel performance data and irradiated fuel samples for safety testing and PIE for key fuel product and process variants. The data obtained from the AGR-2 irradiation and subsequent PIE and safety testing will further increase the fundamental understanding of the relationship among the fuel fabrication process, as-fabricated fuel properties, normal operation, and potential accident condition performance.

3.2.3.3 Fission Product Transport Experiments (AGR-3/4). This multi-monitored capsule test train was a combination of the AGR-3 and AGR-4 experiments originally planned as separate irradiations in large B positions but were combined and placed in the NEFT position in ATR, as also shown in Figure 2. This test train included compacts containing AGR-1 "driver" fuel particles and also seeded with 20 DTF fuel particles, each within rings of graphitic material. DTF fuel particles for use in fission product transport testing consisted of reference kernels with only a ~20-µm-thick pyrocarbon seal coating that was intended to fail as designed during irradiation and provided known fission product source terms. The sweep gas not only contained a mixture of helium and neon necessary to provide thermal control of the experiment but also, in one capsule, gaseous impurities (CO, H_2O) typically found in the primary circuit helium of HTGRs. This allowed for assessing the effect of impurities on intact and DTF fuel performance and subsequent fission product transport. The test train was designed to provide data on fission product diffusivities in fuel kernels and sorptivities and diffusivities in compact matrix and graphite materials for use in upgrading fission product transport models. The AGR-3/4 experiments also have provided irradiated fuel performance data on fission product gas release from failed particles and irradiated fuel samples for safety testing and PIE. The in-pile gas release, PIE, and safety testing data on fission gas and metal release from kernels will be used in developing improved fission product transport models to the extent possible from the experimental results.

3.2.3.4 Fuel Qualification and Fuel Performance Margin Testing Experiments (AGR-5/6/7). This multi-monitored capsule test train is a combination of the AGR-5, AGR-6, and AGR-7 experiments, which were planned originally for separate irradiations in large B positions, similar to AGR-1 and AGR-2, but will be combined and irradiated in the NEFT position in ATR, as shown in Figure 2, the same as AGR-3/4 experiments. The test train will include a single fuel type made using process conditions and product parameters considered to provide the best prospects for successful performance based on process development results and available data[f] from AGR-1 and AGR-2

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b. The decision to proceed with fabrication of qualification test fuel was made based on information available at the time, which included full irradiation of AGR-1 plus PIE, heat-up and fission product metal release data on AGR-1 fuel, as well as in-pile gas release data from AGR-2.

irradiations. This will be the reference fuel design selected for qualification. Variations in capsule conditions (burnup, fast fluence, and temperatures) were established in the irradiation test specifications. The sweep gas will contain helium and neon. This test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE in a sufficient quantity to demonstrate compliance with statistical performance requirements under normal operating and potential accident conditions.

The AGR-7 portion of this test train will include the same fuel type as used in AGR-5/6. The irradiation will test fuel beyond its operating temperature envelope so that some measurable level of fuel failure is expected to occur (margin test). The margin test will provide fuel performance data and irradiated fuel samples for PIE and post-irradiation heat-up testing in sufficient quantity to demonstrate the capability of the fuel to withstand conditions beyond AGR-5/6 normal operating conditions in support of plant design and licensing. The sweep gas will be similar to that used in AGR-5/6.

3.3 PIE and Safety Testing

This program element assesses the performance of irradiated TRISO particle fuel during irradiation and under potential accident conditions. PIE and safety (heat-up) testing are strongly interwoven, because many of the PIE procedures applied to fuel samples following irradiation are also applied to fuel following safety testing. Fuel performance evaluation focuses on quantifying the level of fission product release from the fuel particles and compacts, and on characterizing the condition of kernels and coatings to determine the effect that irradiation or post-irradiation heat-up has on particle microstructure. This work will support the future fuel manufacturing effort by providing feedback on the performance of kernels, coatings, and compacts under varying conditions. Data from PIE and safety testing, in conjunction with in-reactor measurements (primarily fission gas release-rate-to-birth-rate [R/B] ratios), are necessary to demonstrate that the quality and performance of the fuel system meet the reactor design requirements. Thus, data from this activity will likely constitute a primary element of the licensee’s fuel qualification submittal to the NRC to obtain an operating license for the first plant.

3.3.1 Goals, Assumptions, and Objectives

The goals, assumptions, and objectives of PIE and safety testing activities are as follows.

3.3.1.1 Goals

- Collect relevant fuel PIE and safety testing data as a function of temperature, burnup, fast fluence, and coolant chemistry for developing and validating fuel performance and fission product transport models, and to demonstrate acceptable fuel behavior under normal operating and potential accident conditions.
- Cooperate with other DOE-NE programs, and use international collaboration as much as possible to resolve key design data needs and minimize duplication of effort.

3.3.1.2 Assumptions

- HTGRs will be designed such that the radionuclides are substantially retained within the coated fuel particles during normal operation and all design basis accidents.
- Water or moisture ingress accidents are mitigated to have only moderate ingress flow rates rather than core flooding.
- Air ingress accidents are to be considered.

- DOE-NE will implement the requisite cooperative agreements to facilitate cooperation with other DOE-NE programs as well as international cooperation.

3.3.1.3 Objectives

- Confirm that fuel performance under normal operating and potential accident conditions can be predicted within the prescribed accuracy limits.
- Collect data to allow validation of the design methods used to predict fuel performance to prescribed accuracy limits in a manner acceptable to regulators and stakeholders.
- Improve understanding of TRISO fuel behavior based on observed and measured phenomena that affect fuel performance and fission product release.

3.3.2 Scope of PIE and Safety Testing

In most cases, the major PIE and safety testing design data needs are sufficiently well known and lead directly to the measurements or tests to be performed to satisfy them. In some cases, development of a new measurement technique is required to satisfy a specific design data need, which leads to a task to develop or apply that new technique.

HTGR fuel has been examined and tested at ORNL since the 1960s. The ORNL hot cells and Core Conduction Cooldown Test Facility (CCCTF) have a full range of capabilities to support the required examinations and safety testing. INL hot cells have also been used to examine a wide variety of irradiated fuels for many years, including TRISO-coated lithium-target particles for tritium production in the NPR program. The relevant facilities at INL and ORNL were operating and functional at the beginning of this program, and both laboratories had development staff capable of designing, procuring, and installing the equipment, and developing the protocols for new or additional examination methods required for the AGR Fuel program.

Equipment and enhancements were added to INL capabilities, including the following:

- Test train and component disassembly tools
- Remotely operable metrology equipment
- Fuel accident condition simulator (FACS) furnace in the Hot Fuel Examination Facility (HFEF) hot cell
- Various safety-testing and analytical equipment for the test trains, capsules, and components of the experiments
- New feed-throughs and cabling to the FACS furnace for remote operation and data collection
- Modifications to the HFEF precision gamma scanner, including installation of a Compton shield
- Replacement of existing camera equipment in the HFEF main cell with a digital camera
- Electron probe micro analyzer for use in advanced microscopy in the Irradiated Materials Characterization Laboratory at MFC
- Advanced focused-ion beam instrument with scanning electron microscopy capabilities for installation in the Irradiated Materials Characterization Laboratory.

Equipment and enhancements were also added to ORNL capabilities, including the following:

- Second-generation advanced irradiated microsphere gamma analyzer for gamma counting of individual particles
- Particle micro-manipulator in the hot cell cubicle that also houses the irradiated microsphere gamma analyzer
- Addition of an interlock system to the CCCTF furnace system, allowing hot exchange of the cold finger deposition plates for time-dependent data collection for fission product release
- Upgraded CCCTF sweep gas-monitoring system
- Upgraded gamma counting hardware
- Improved liquid nitrogen supply system
- Additional scanning electron microscopy capability
- Deconsolidation/ LBL system
- Addition of a Struers MiniMet polishing system to materialography capability
- Addition of a customized shielded sample enclosure to the x-ray tomography system.

Procedures and instructions were developed for, and personnel were trained on, equipment and processes to meet NQA-1 requirements at INL and ORNL prior to their use.

The tasks associated with PIE and safety testing are discussed below. As noted earlier, some PIE tasks may not be required depending on results as the activity proceeds, but costs are based on currently planned PIE and safety testing to provide the best estimate for program planning purposes. Determining the required tasks for a particular test train occurs during preparation of the PIE and safety test plan. Adjustments to the plan are made throughout the PIE and safety testing campaign based on results obtained during earlier examinations and testing and on budgetary considerations. Whether a full range of examinations is required for fuel irradiated under the AGR Fuel program depends on many factors, including the defective fuel fraction measured during manufacturing and the in-pile R/B measurements. If the fuel manufacturing effort is successful, the fuel should have few, if any, defective particles (a fraction of exposed uranium $<10^{-4}$) and a low in-pile R/B ($<10^{-6}$). PIE will primarily address metallic fission product release fractions, distributions within the fuel and graphite, and coating layer behavior but will also utilize the available capabilities to locate and examine any failed fuel coatings within particles. Cost estimates and tentative schedules for conducting PIE and safety testing are provided in Section 4.

3.3.2.1 General PIE and Safety Testing, Assessment, and Facility Preparation. The following subsections discuss the required preparations to conduct PIE and safety test activities, list the test trains, and briefly summarize the PIE and safety test objectives for each test train. The subsections discuss in detail the PIE and safety test tasks and identify the subset of tasks to be performed for each of the test trains.

General PIE needs of the program involve:

- Transport of the irradiated and sized test train from ATR to MFC (requires one or two shipments to complete based on test train length and available shipping package)
- Test train receipt, offloading, inspection, and handling at MFC
- Gamma scanning of the intact test train at MFC

- Test train disassembly at MFC
- Separation and disassembly of irradiated capsules at MFC
- Metrology of irradiated compacts and components at MFC
- Nondestructive examination of fuel compacts and capsule components at MFC
- Measurement of fission product inventory in the irradiated capsule components at MFC
- Transport of selected irradiated compacts to ORNL for safety testing and PIE
- Detailed destructive analysis of the irradiated fuel compacts at INL and ORNL
- Extensive microanalysis and evaluation of the level of fission product retention in the irradiated particles and compacts at INL and ORNL.

Safety testing needs involve:

- Measurement of fission product release from fuel compacts at elevated temperatures using the FACS and CCCTF furnaces at INL and ORNL, respectively
- Destructive analysis of the fuel compacts following the safety (heat-up) tests at INL and ORNL
- Heating of graphite and matrix ring materials to induce fission product migration and aid in quantifying transport properties primarily at INL (AGR-3/4 experiment only).

Many of the facilities and equipment required for these tasks were in place at the beginning of the AGR Fuel program, although restoration, upgrading, improvements, and new capabilities have been necessary for some tasks. Many of these tasks have been conducted in the past, and the available experience has been factored into the reestablishment of these capabilities.

3.3.2.2 AGR Experiment PIE Preparations. For each of the first three test trains (i.e., AGR-1, AGR-2, and AGR-3/4), the four tasks described below have been initiated to address the needs listed above. AGR-1 and AGR-2 test trains have very similar designs, so equipment developed for AGR-1 disassembly and metrology was used as much as possible for AGR-2 disassembly and metrology. However, there were lessons learned during AGR-1 PIE processes and equipment that have been and continue to be applied to AGR-2 PIE to improve the capabilities, simplify tasks where practical, and reduce costs. The AGR-3/4 test train has a much different design, so new disassembly tooling was designed and fabricated, and lessons learned from earlier PIE efforts were incorporated.

PIE Site Task

Prior to initiating AGR-1 PIE, the capabilities of candidate facilities, existing and new, for performing the separate PIE tasks were reviewed, and new equipment was developed to perform new tasks for which no capability previously existed. This task also involved determining how these facilities might be integrated and consideration of the implications of transport and time delays that might impact analysis, cost, and schedule.

PIE Preparation Task

The selected facilities at INL and ORNL have been prepared for PIE and safety testing. The PIE and safety testing capabilities were inventoried, and new equipment was developed. Procedures and instructions for operations personnel to follow during the performance of each of the PIE tasks were developed. The necessary environmental, safety, and health documentation was prepared to protect workers, the public, and the environment.

The facilities and apparatus required to perform the PIE and safety testing were made ready and upgraded, as needed, to meet performance expectations. Resources were allocated within the AGR Fuel program to develop detailed cost estimates for the requisite upgrades and new capabilities for each experiment.

Generally, the nominal time to complete PIE and safety testing of an irradiated test train is about 4 years, assuming that facilities and personnel are available. AGR-1 PIE and safety testing have taken place over about 5 years because of various issues that arose during the initial setup and performance of the various activities and the learning curves associated with them. The overall AGR Fuel program irradiation schedule has resulted in AGR-2 and AGR-3/4 PIE and safety testing commencing within about 9 months of each other. The sharing of PIE and safety test work at two sites (INL and ORNL) is necessary to handle the workload. This is most pressing for complex, time-consuming tasks such as the safety tests, which involve high-temperature heat-up for extended periods followed by detailed fuel examination. The AGR-2 PIE and safety testing are being performed primarily at ORNL, while AGR-3/4 PIE is being performed primarily at INL. These work activities are split because of additional capabilities at ORNL for destructive examination and particle analysis; a lack of AGR-3/4 sample transport methods from INL to ORNL for the graphite, matrix rings, and fuel bodies; and time constraints to complete PIE on both experiments within a reasonable time.

Helium/Air/Steam Ingress Safety Testing Development Task

The CCCTF fuel heat-up facility (furnace) at ORNL is capable of 1800°C in a helium atmosphere. A second furnace, the FACS furnace, located at INL is capable of heat-up to 1800°C in helium. AGR-1 safety testing has been done in pure helium to provide a data set for comparison with the extensive historical database of German fuel safety tests, which were performed in a similar atmosphere. A new furnace or fuel heat-up facility is being developed to extend the chemical environment capabilities to temperatures as high as 1600°C in oxidizing atmospheres typical of air- and moisture-ingress events. To be prepared for these heat-up tests to support scheduled AGR-5/6/7 PIE activities, the AGR program began developing this capability in early FY 2016.

Fuel Compact Re-Irradiation Equipment Development Task

A method to re-irradiate loose fuel particles in a reactor so the release of short-lived radioisotopes, including ^{131}I with a half-life of 8 days and ^{133}Xe with a half-life of 5 days, can be measured in safety tests is being pursued. The Neutron Radiography Reactor (NRAD) at MFC is the planned location for the re-irradiation.

3.3.2.3 PIE and Safety Testing Scope of Activities. Test train PIE and safety testing are composed of several tasks selected from various options. Some of these tasks will be conducted in parallel, while others must be conducted sequentially. The actual grouping and relationships of the tasks are detailed in a specific experimental PIE plan prepared for each AGR experiment. For planning purposes, it is assumed that AGR-2 and AGR-3/4 PIE and safety testing will take about 5 years to complete, assuming no further restrictions on facility use or resources. The following tasks outline the options that are likely to be available for PIE and safety testing. The planned tasks expected to be performed for each particular test train are shown in Table 2.

The PIE and safety testing tasks are being integrated with other activities so that tasks can be conducted efficiently. The primary goal is to ensure that the needed measurements and tests are accomplished with the required accuracy. If this is impossible, the program needs early notification so alternative actions can be taken. In particular, some data may prove to be very expensive or time-consuming to collect, and different approaches to modeling or fuel qualification may have to be explored.

The following list provides a brief description of the PIE tasks outlined in Table 2. Because of shipping responsibility boundaries, the shipment of test trains from ATR to MFC is covered by an MFC nuclear facility manager, but test train shipment is not considered a PIE task for the purposes of this description.

- *PIE TASK-1: Test train receipt and visual inspection.* The transfer and nuclear accountability documentation will be completed, and the HFEF truck lock will be prepared for the receipt of the GE-2000 cask or other approved shipping configuration containing the test train. The shipment will be transported from ATR to the HFEF truck lock. The test train will be removed from the cask in the truck lock and moved into the HFEF hot cell where photo-visual examination of the test train will be conducted. The AGR-1 and AGR-2 test trains were shipped in the GE-2000 cask as individual shipments. The length of the AGR-3/4 test train and the internal dimensions of the GE-2000 cask required that it be shipped in two sections and two shipments. It is anticipated that this shipping configuration will also be required for the AGR-5/6/7 test train if the GE-2000 cask or similar shipping system is used.
- *PIE TASK-2: Test train nondestructive examination*. The intact test train will be analyzed in the HFEF main cell using the precision gamma scanner for a high-resolution gamma scan in the axial direction to help verify the position of the test train's internal components. Neutron radiography in NRAD may also be used to perform nondestructive examination of test train's internal components.
- *PIE TASK-3: Test train and capsule disassembly*. The test train and capsules will be disassembled in the HFEF hot cell using in-cell disassembly equipment, tools, and jigs to remove the fuel compacts and internal components of experimental value.
- *PIE TASK-4: Component metrology*. The fuel compacts and internal capsule components will be visually and dimensionally inspected in the HFEF hot cell. After completion of this task for each of the AGR-1[33], AGR-2[34], and AGR-3/4 experiments, a "first look report"[35] has been issued with extensive photographs and descriptions of the initial findings regarding the physical appearance of the test trains and components.
- *PIE TASK-5: Compact shipments to ORNL.* Selected compacts of interest will be packaged and shipped from INL to ORNL for concurrent PIE and safety testing. Shipments of compacts to ORNL are planned to be made in approved shipping packages by a commercial carrier. Twenty AGR-1 compacts were shipped to ORNL for PIE and safety testing. At least 20 AGR-2 compacts are planned to be shipped from INL to ORNL in two shipments of four AGR-2 compacts each per year in FYs 2015, 2016, and 2017, with 16 compacts shipped through FY 2016. There are no current plans to ship AGR-3/4 compacts to ORNL for PIE or safety testing. AGR-5/6/7 compact shipping plans will likely be similar to AGR-2 depending on available approved shipping packages.
- *PIE TASK-6: Graphite fuel holder and graphite/matrix ring gamma scanning.* Empty graphite fuel holders from AGR-1 and AGR-2 have been gamma scanned to quantify total inventory and identify potential hot spots from fission product release. AGR-3/4 graphite and matrix rings are being gamma scanned to determine the inventory and distribution of fission products retained in the rings. AGR-5/6/7 graphite holders will be scanned for fission products to quantify total inventory and identify any locations with elevated activity that may be indicative of compacts containing particles with failed SiC. AGR-5/6/7 graphite holders as well as graphite and matrix rings from AGR-3/4 will be scanned for fission products. If detected, the fission product distribution will be mapped to determine the location of hot spots.

- *PIE TASK-7: Fuel compact gamma scanning.* Fuel compacts will be characterized with gamma spectroscopy to determine inventories of key fission products and measure fuel burnup.
- *PIE TASK-8: Melt and flux wire analysis.* Melt and flux wires will be removed from the graphite holders and analyzed to determine neutron flux levels and possible indications of high temperatures. The wires will be analyzed at either Pacific Northwest National Laboratory (the original manufacturer of the melt and flux wire packages) or INL.
- *PIE TASK-9: TC analysis.* Selected TCs may be examined to identify any chemical interactions that could have implications for fuel coating interactions and to aid in test train design. Because this activity has a low priority relative to other fuel characterization activities, and because successful analysis of these very fine parts is significantly complicated by working in the hot cell environment, analysis of the TCs has not been performed on the AGR-1, AGR-2, and AGR-3/4 experiments. No decision has been made yet on AGR-5/6/7 and whether this analysis will be performed in the future.
- *PIE TASK-10: Properties of irradiated materials specimens*. Properties (thermal, physical, and mechanical) may be measured on samples of irradiated materials. This may be accomplished by irradiating non-fissile surrogate materials (e.g., pure AGR fuel matrix or Zr-TRISO in compacts) in Advanced Graphite Creep (AGC) experiment graphite capsules for cost, schedule, and practical considerations; matrix-only specimens have been irradiated in the AGC-2 experiment for 13 months, and PIE has been performed on them. Additional matrix only specimens will be irradiated in the AGC-4 experiment for about 2 years, concluding in FY 2018 with PIE to be performed afterward as part of the AGC program.
- *PIE TASK-11: Capsule deposited fission products.* The interior metal surfaces of each capsule will be analyzed for the presence of fission products that were released from the fuel during irradiation (except in the case of AGR-3/4, where the cold graphite sink ring is expected to act as a barrier to fission product migration to the steel capsule shell). A quantitative analysis will be performed on other capsule components (including the graphite and matrix components) from each capsule so that a mass-balance of fission products (including non-gamma-emitting fission products) released from the fuel can be determined.
- *PIE TASK-12: Radionuclide transport in irradiated specimens.* Radionuclide content and gradients in irradiated AGR-3/4 matrix material and graphite specimens will be measured using appropriately established techniques such as beta and gamma spectrometry and physical sampling and analysis.
- *PIE TASK 13: Micro-scale analyses of fuel compacts.* Selected compacts from irradiation and after safety testing will be analyzed in cross section at the microscopic scale to assess localized effects of irradiation and post-irradiation heating on the compact matrix and embedded fuel particles. This has been completed for AGR-1 compacts, is under way for AGR-2 compacts at the time of this writing, and is planned to be performed on AGR-5/6/7 compacts.
- *PIE TASK-14: Compact deconsolidation.* Selected compacts from each of the experiments will be deconsolidated to free individual fuel particles from the matrix binder as a precursor to the LBL process and to provide loose fuel particles for other PIE tasks.
- *PIE TASK-15: Compact LBL.* The standard procedure is to perform an initial acid leach on deconsolidated compacts, particles, and matrix to dissolve uranium and fission products in the matrix and exposed kernels. Based on AGR-1 PIE experience, an alternative approach that may be employed is to perform compact deconsolidation only, with no subsequent elevated temperature pre-burn acid leaches, in order to avoid complete destruction of particles and acid digestion of the kernels. This could be employed in cases where compacts have been identified as potentially containing particles

with SiC layer failures, and will increase the likelihood of harvesting the particles intact for gamma analysis and subsequent microanalysis The particles and matrix debris will be exposed to air at elevated temperatures (750°C) to oxidize matrix and PYC material not protected by intact SiC coatings. A post-burn leach will then be performed to dissolve any additional fission products that were present in the matrix debris or the OPyC layer and to dissolve uranium and fission products exposed by the burn step. These tasks are usually combined with *PIE TASK-14* above.

- *PIE TASK-16: Particle inspection and sorting.* Intact particles from deconsolidation and/or LBL will be optically examined with sufficient magnification to provide an indication of the condition of the particles, including coating damage, if any.
- *PIE TASK-17: Burnup measurement.* The primary means of burnup measurement will be activity ratios determined from the compact gamma scans in *PIE TASK-7*. Destructive isotopic analysis methods will be used on particles from selected compacts as a benchmark to compare with the burnup determinations from the gamma scanning data.
- *PIE TASK-18: Irradiated microsphere gamma analysis.* Individual particles from each of the experiments will be gamma counted to quantify the inventories of selected fission products. The data will be used to gauge the relative fission product retention in each of the analyzed particles and can be used to screen for failed particles based on radionuclide inventories before performing other analyses.
- *PIE TASK-19: Microanalysis of fuel particles.* Particles identified in the previous tasks will be prepared in cross section for individual examination, including optical microscopy, scanning electron microscopy, transmission electron microscopy, high-resolution transmission electron microscopy, scanning transmission electron microscopy, and chemical analysis using energy dispersive spectroscopy and wavelength dispersive spectroscopy. This task may also include analysis of intact particles using x-ray tomographic methods. Additional advanced microanalysis methods that may be used include electron energy loss spectroscopy in conjunction with the scanning transmission electron microscopy, electron backscatter diffraction, and atom probe tomography.
- *PIE TASK-20: Safety testing – re-irradiation.* Selected compacts or particle samples will be re-irradiated before safety testing, primarily to generate short-lived fission products, including ^{131}I and ^{133}Xe, so I and Xe release during safety testing can be measured. This will most likely take place in NRAD at MFC.
- *PIE TASK-21: Safety testing.* Selected compacts will undergo heat-up tests in helium at peak temperatures of 1400 to 1800°C for planned durations of approximately 300 consecutive hours. Both isothermal and variable temperature profiles will be used. Gaseous fission product release will be measured continuously during the test, and condensable fission product release will be measured by analysis of condensate surfaces within the furnace that are periodically replaced and analyzed for deposited isotopes. A separate fuel safety testing capability will be developed to extend the chemical environment capabilities to temperatures up to approximately 1600°C in an oxidizing atmosphere typical of air- and moisture-ingress events. This capability will be used to test AGR-5/6/7 fuel compacts. Additional tests in oxidizing atmospheres will be performed using archived, irradiated AGR-3/4 fuel compacts, focused specifically on the effect of air and moisture on fission product release from exposed kernels.

- *PIE TASK-22: Graphite and matrix heating tests.* Irradiated graphite and matrix specimens, with fission products deposited in them during irradiation, will be heated in a variety of atmospheres (potentially including dry helium and helium with various concentrations of air or moisture) while measuring fission product release. Tests in helium can also be used to help derive diffusion coefficients for various fission products from the rings.

- *PIE TASK-23: Archiving and waste handling*. Some fuel specimens in various configurations (kernels, TRISO particle fuel, and compacts) will be collected and placed into archives at INL and ORNL for further research or historical purposes. Residual materials not chosen for archival storage will be handled as waste. Collecting, packaging, and disposing of irradiated fuel specimens and associated waste generated during AGR PIE will take place at ORNL and INL. The type of waste involved will determine its need for treatment or its disposition path.

- *PIE TASK-24: Reporting*. Researchers will disseminate the findings, results, and lessons learned from the PIE task in formal and informal reports, presentations, and publications. Also, there will be support for program requests for specific information, clarifications, and impact assessments.

3.3.3 Test-Train-Specific PIE and Safety Testing

A preliminary assessment of the applicability of the detailed PIE and safety testing tasks defined above to the individual irradiation test trains, based on the objectives of each test train, resulted in the task assignments shown in Table 2. The objectives of the PIE and safety testing of each test train are summarized in Subsections 3.3.3.1 through 3.3.3.4. Estimated costs and tentative schedules for PIE and safety testing of each test train are provided in Section 4.

3.3.3.1 AGR-1: Shakedown Test; PIE of Test Train and Early Fuel. As previously noted, the initially planned purpose of AGR-1, the first test train to undergo irradiation, PIE, and safety testing, was to gain experience with multi-monitored capsule test train design, fabrication, and operation, and to reduce the chances of test train or capsule failures in subsequent test trains. An additional purpose was to reestablish, develop, and shake down PIE and safety testing equipment and methods to be used for later experiment irradiations.

However, the scope of AGR-1 PIE was substantially expanded to:

- Provide extensive data on fuel performance under irradiation and simulated accident testing to support specification of the fuel to be qualified in later experiment irradiation test trains

- Support early HTGR pre-licensing interactions with the NRC

- Develop a quantitative understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance.

The specific PIE and safety testing tasks performed on this test train are identified in Table 2. The individual task scopes are summarized in Subsection 3.3.2.3.

3.3.3.2 AGR-2: PIE of Fuel Performance Test Train. The AGR-2 PIE and safety testing will provide irradiated fuel performance data beyond the online R/B measurements for UCO and UO_2 fuel types fabricated in the larger production-scale (6-in.) coater, as discussed in Subsection 3.2.3.2. The PIE and safety testing also support development of a fundamental understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance. The specific PIE tasks and safety test tasks performed so far or planned to be performed on this test train are identified in Table 2.

Table 2. Test train PIE tasks.

Task Number	Task	AGR-1	AGR-2	AGR-3/4	AGR-5/6/7
PIE TASK-1	Test train receipt and visual inspection	X	X	X	X
PIE TASK-2	Test train nondestructive examination	X	X	X	X
PIE TASK-3	Test train and capsule disassembly	X	X	X	X
PIE TASK-4	Component metrology	X	X	X	X
PIE TASK-5	Compact shipments to ORNL	X	X	X	X
PIE TASK-6	Graphite holder gamma scanning	X	X	X	X
PIE TASK-7	Fuel compact gamma scanning	X	X	X	X
PIE TASK-8	Melt and flux wire analysis	X	X	X	X
PIE TASK-9	TC analysis		TBD	TBD	TBD
PIE TASK-10	Properties of irradiated material specimens				X
PIE TASK-11	Capsule deposited fission products	X	X	X	X
PIE TASK-12	Radionuclide transport in irradiated specimens			X	
PIE TASK-13	Microanalysis of fuel compacts	X	X		X
PIE TASK-14	Compact deconsolidation	X	X	X	X
PIE TASK-15	Compact LBL	X	X	X	X
PIE TASK-16	Particle inspection and sorting	X	X	X	X
PIE TASK-17	Burnup measurement	X	X	X	X
PIE TASK-18	Irradiated microsphere gamma analysis	X	X	X	X
PIE TASK-19	Micro-scale analysis of fuel particles	X	X	X	X
PIE TASK-20	Safety testing – particle re-irradiation		X	X	X
PIE TASK-21	Safety testing – heat-up tests	X	X	X	X
PIE TASK-22	Graphite and matrix heating tests			X	
PIE TASK-23	Archiving and waste handling	X	X	X	X
PIE TASK-24	Reporting	X	X	X	X

3.3.3.3 AGR-3/4: PIE of Fission Product Transport Test Train. The AGR-3/4 PIE and safety testing will provide data to support calculation of fission product diffusivities in fuel kernels and coated particles, and fission product diffusivities and sorptivities in fuel compact matrix and graphite for use in upgrading fission product transport models and codes. This PIE will focus on measurements of fission product inventories and concentration profiles in the graphitic components with a focus on a full mass balance to support fission product transport model development. However, the PIE activities will also involve heating fuel compacts and irradiated graphite and matrix materials in a variety of test atmospheres (potentially including dry helium, air, and moisture) while measuring fission product release. The specific PIE and safety test tasks performed so far, or planned to be performed, on this test train are identified in Table 2.

3.3.3.4 AGR-5/6/7: PIE of Fuel Qualification and Fuel Performance Limits Test Train.

This is now planned to include three experiments, i.e., AGR-5/6/7, in a single test train. The AGR-5/6/7 PIE and safety testing will document fuel integrity and safety test performance to demonstrate compliance with statistical performance requirements under normal operating and potential accident conditions. The primary interest is to verify successful fuel performance. This PIE makes heavy use of the fuel heat-up capabilities. The AGR-7 PIE measures the capability of the selected fuel to withstand irradiation and potential accident conditions beyond the conditions in AGR-5/6 in support of plant design and licensing. The specific PIE and safety test tasks planned for this test train are identified in Table 2.

3.3.4 Moisture Ingress

The current interest in using high-temperature steam for process heat applications has led to including a steam generator in the primary helium coolant system in recent evaluations of design options for the HTGR. This design option brings with it the risk of steam generator tube leaks, resulting in moisture ingress into the primary coolant system of the HTGR. Thus, there is an increased need to address the effects of moisture ingress on fuel behavior and fission product release from the core and transport in the primary coolant system and reactor building.

A review of the considerable experimental and analytical work on moisture-ingress accidents has been published for the AGR Fuel program.[36] This review treats analyses of a range of moisture ingress accidents for the MHTGR with a steam generator in the primary system, as documented in the MHTGR preliminary safety information document.[37] The review also addresses experiments on moisture ingress, both in-pile and out-of-pile post-irradiation safety tests, and extensive analysis of the experimental data. Although detailed models of fuel hydrolysis have been developed for the purposes of assessing the consequences of moisture-ingress accidents on fission gas release from the fuel, it may be sufficient to know that (1) hydrolysis affects only fuel particles with exposed kernels; (2) gas release is dominated by the release of stored gas; (3) release of stored gas is independent of the type of gas (Xe or Kr) and the isotope of the gas; and (4) the fractional quantity released is independent of the fuel chemistry, UCO or UO_2.

The principal results of the experiments on fuel hydrolysis can be reduced to a logarithmic plot of fractional release of stored gas as a function of partial pressure of water vapor, where data from both in-pile and out-of-pile experiments fall on the same curve.[36] Recent re-examination of the analyses of moisture-ingress accidents documented in the MHTGR preliminary safety information document[38] indicates that the reactor is scrammed quickly (within 8 to 22 seconds) after initiation of moisture ingress, whereas minutes to hours are required to release fission gas from the fuel in the experiments. While the coolant flow in experiments tends to be much slower than in an operating gas-cooled reactor, the experimental results indicate that the vast majority of fission gases released by fuel hydrolysis in a moisture-ingress accident will be generated after reactor shutdown.

Experiments to measure the oxidation of fuel-element graphite and fuel-element matrix (both of HTGR specifications) under moisture-ingress conditions are described in a research plan for moisture and air ingress.[21] Two types of out-of-pile experiments were proposed for the AGR program. One used irradiated-fuel compacts containing a known fraction of DTF UCO particles, with some of the compacts enclosed in a graphite body; the other used unirradiated fuel compacts, which contained intact TRISO-coated surrogate (ZrO_2) fuel particles, in a mockup of a graphite fuel element. In the experiment with irradiated fuel compacts, the release of metallic and gaseous fission products will be measured as a function of time at temperature. A source for irradiated compacts containing DTF fuel particles, some enclosed in a graphite body, is from the AGR-3/4 experiment. For unirradiated surrogate material testing, the program is focusing in the near term on testing using (1) pure matrix specimens and (2) surrogate TRISO particles with no OPyC layer, such that the SiC layer is exposed. These will be used to separately assess the reaction rates of graphitic matrix and SiC in a variety of atmospheres, whereas tests on whole surrogate compacts inside graphite sleeves would make evaluation of reaction rates of the individual components impractical. Follow-on testing on compacts as suggested in Reference 21 may also be pursued, which may provide an estimate of oxygen partial pressures experienced at the TRISO particle SiC layer surface, resulting from oxidant flow in the reactor core graphite cooling channels. Although water partial pressure as high as 3.5 atm could be expected in MHTGR accident scenarios,[39] calculations show that above about 0.3 atm partial pressure of water vapor, the rate of H-451 graphite oxidation is independent of steam pressure.[40] Testing should determine the water partial pressure at which the graphite and matrix oxidation rates saturate. The test parameters should span the transition between passive (formation of stable SiO_2) or active (formation of volatile SiO) oxidation of SiC as predicted by thermodynamic studies of the SiC-C-H_2O system.[41]

Another aspect of moisture relates to impurities in the helium coolant gas. The impurity levels in the helium coolant gas are much lower than the moisture levels in the ingress experiments discussed above. Testing with typical levels of impurities was included in the AGR-3/4 experiments. Design data needs for MHTGRs with steam generators, going back to 1987,[42] specify in-pile testing with impure helium containing 12.6 Pa of H_2O and CO_2 and 31.5 Pa of CO. More recently, an analysis of impurities in operating gas-cooled reactors, including historical reactors such as Dragon, Peach Bottom-1, AVR, Fort St. Vrain and the currently operating high-temperature engineering test reactor (HTTR), suggests that in a modern gas-cooled reactor, impurities of somewhat less than 1 Pa for H_2O and CO_2 and in the range of 1 to 10 Pa for CO and H_2 could be expected.[37] Considering the massive amount of graphite and the high operating temperatures in the core of an HTGR, the two key impurities that will influence the oxygen potential are CO and H_2O. Therefore, the AGR-3/4 irradiation test contained a capsule with representative quantities of CO and H_2 (about 5 Pa) and H_2O (about 1 Pa) in the inlet helium coolant/neon purge gas. However, experiment data and further analysis indicated that the impurities reacted with capsule materials in the inlet section before reaching the fueled section.

3.3.5 Air Ingress

Air ingress into the core of an HTGR may occur following a depressurization accident. The severity of the event depends on break size, break location, and design of the reactor cavity, all of which influence the ability of air to enter the core via natural circulation, stratified flow, or molecular diffusion. In both a prismatic and a pebble-bed HTGR, a graphite or matrix thickness of about 5 mm must be permeated before air or reaction products can have access to a fuel compact (prismatic design) or fueled region of a spherical fuel element (pebble-bed design).

Ingress of air into the reactor core following coolant depressurization has been comprehensively studied both analytically and experimentally.[43,44,45] A major finding has been that the onset of natural circulation of air through the core is rapid, on the order of 100 to 500 seconds, following complete depressurization. Natural circulation begins relatively sooner for a larger break, whereas for a smaller break, the depressurization takes many hours before the system is completely depressurized and natural circulation can begin. The flow rate through the core under natural circulation is in the range of 0.1 to 0.2 m/second for all break sizes and locations, because the principal driving force is the temperature difference between the core and the coolant riser. These values are expected to hold for either a prismatic or pebble-bed core. Note that these values do not take into account any mitigating safety systems but are useful starting points for planning experiments.

As discussed in Reference 46, graphite oxidation is expected to be a small contributor to the overall safety consequences for a beyond-design-basis accident with air ingress. Analytical results also show that the safety consequences are not very sensitive to the time period required to establish natural convection air flow through the reactor vessel for time periods ranging from 0 to 4 days.[46]

It should be noted that immediately following a larger break depressurization event (depressurization times ranging from a few seconds to several minutes), the atmospheres within the reactor building cavities that house the reactor vessel, steam generator, or heat exchanger vessel and other equipment will consist mostly of inert helium. This condition is a result of the reactor building designed as a vented, low-pressure containment. The reactor building will vent its atmosphere through passive louvers until it returns to its low-pressure design point. The reactor building atmosphere will be mostly helium at this point, because the helium pressure boundary on a molar basis typically contains 4 to 5 times the gas content of the reactor building cavities. A mostly inert helium atmosphere within the reactor building at the end of a major depressurization event could greatly delay the onset of any significant air ingress into the reactor vessel. The relevant phenomena that control the reactor building atmosphere during depressurization events are being investigated both analytically and experimentally under a separate DOE-funded program.

Measurements of the oxidation of IG-110 graphite (the Japanese standard graphite used in the HTTR core) in air indicate that the oxidation rate increases with increasing oxygen concentration and saturates at temperatures of about 1200°C.[43]

Results of experiments using unirradiated compacts in Japan and irradiated fuel spheres in Germany have been reported.[47] The Japanese results of weight change as a function of time indicate that oxidation of carbonaceous materials in an unirradiated fuel compact is complete after 20 hours in flowing air at 1400°C, revealing the SiC layer of the particles. After this duration, the particle failure fraction was determined to be 6.9×10^{-4}. After 54 hours in flowing air at 900°C, a particle failure fraction of 1.2×10^{-3} was measured. German results on irradiated spherical fuel elements (burnup of about 9% FIMA) indicated a particle failure fraction of 2.4×10^{-4} after 410 hours in flowing air at 1300°C, 7.3×10^{-4} after 70 hours at 1400°C, and 1.2×10^{-3} after 140 hours at 1400°C, although Schenk data reported in *Fuel Performance and Fission Product Behavior in Gas Cooled Reactors*[44] gave higher failure fractions. These limited results suggest that significant fuel failure can be expected after tens of hours in flowing air at temperatures in the range of 1300 to 1400°C. However, the experimental conditions for this work, particularly for the studies of fuel compact oxidation in Japan and of fuel sphere oxidation in Germany, may not be representative of the actual conditions that would exist during air-ingress events. The supply of air would be limited, and the graphite structural materials in the core, in particular the fuel-element graphite of the prismatic fuel elements, would limit the amount of air that could reach the fuel compacts.

Experiments to measure the oxidation of fuel-element graphite and fuel-element matrix (both of HTGR specification) under more representative air-ingress conditions are described in the research plan for moisture and air ingress.[21] The same two types of out-of-pile experiments described for water ingress were proposed for air ingress tests; the two types of out-of-pile experiments are (1) compacts with ZrO_2 TRISO particle fuel in a graphite fuel element mockup and (2) post-irradiation heat-up of UCO fuel compacts, some of which contain DTF particles and some of which are enclosed in a graphite body.

Irradiated AGR-3/4 compacts with DTF particles can be tested in a manner similar to that described in the previous subsection. As described above for moisture testing, testing of unirradiated surrogate material is focusing in the near term on testing using (1) pure matrix specimens and (2) surrogate TRISO particles with no OPyC layer, such that the SiC layer is exposed. These will be used to separately assess the reaction rates of graphitic matrix and SiC in a variety of atmospheres. Follow-on testing of whole surrogate compacts in graphite sleeves, as described in Reference 21, may also be pursued. It is recommended that the gas flow rate in the experiments be in the range of 0.1 to 0.2 m/second. The nominal temperature range is 1000 to 1600°C, and the range of the fraction of air in helium will be determined based on calculations with codes (e.g., OXIDE[48] and GAMMA) and initial experiment results. The test parameters should span the transition between passive to active SiC oxidation, as predicted by thermodynamic studies of the SiC-C-O_2 system.[49]

3.4 Fuel Performance Modeling

A key product of the AGR Fuel program is the development of validated fuel performance models. As discussed here, fuel performance modeling addresses the structural, thermal, and chemical processes that can lead to coated-particle failures. The modeling considers the effects of fission product chemical interactions with the coatings, which can lead to degradation of the coated-particle properties. Fission product release from the particles and transport within the fuel-compact matrix and fuel-element graphite are also modeled. Many groups have attempted to model the performance of coated-particle fuels.[50] These efforts have not resulted in a comprehensive model capable of predicting fuel performance with sufficient accuracy to directly facilitate fuel design or replace the need for comprehensive test data in a licensing application. The most significant reasons the modeling has not yet succeeded are (1) incomplete representative coating property data as a function of irradiation conditions and (2) insufficient understanding of the interactions between phenomena as irradiation proceeds. Thus, the goals are to:

- Develop fuel performance models of coated-particle fuel (either UCO or UO_2) that are more first-principle based and can be used to:
    - Guide current and future particle designs
    - Assist in irradiation and safety test experiment planning
    - Predict observed fuel failures and fission product release
    - Allow more accurate interpolation of fuel performance inside the performance envelope needed for core design assessments and modest extrapolation of fuel performance outside the existing performance envelope when required
- Develop a prioritized list of material properties and constitutive relations needed for accurate modeling of coated-particle fuel under normal and off-normal conditions
- Develop advanced models that take advantage of new methods
- Benchmark these models/codes against United States and international irradiation and safety test experiments, where possible.

The effort by the modeling working group has been focused on improving these crucial areas. Performance modeling is an iterative task. Work began on modeling during the days of the Dragon Project in the 1960s and continued through the 1990s, as documented in the results of an International Atomic Energy Agency (IAEA) Coordinated Research Project on fuel performance and fission product behavior.[44] More recently, another IAEA Coordinated Research Project code-to-code benchmark was conducted with improved models.[51] While useful, currently available models are not adequate for the applications mentioned earlier. Models will continue to evolve throughout the fuel development phase and into the period of commercial fuel manufacturing and power generation. This has been the case with every reactor system deployed for electricity production.

Fuel performance models are used for (1) assisting in developing candidate coated-particle fuel designs; (2) predicting the performance of coated-particle fuel during irradiation testing and post-irradiation heat-up; and (3) calculating fuel performance for HTGR core designs under normal operating and hypothetical accident conditions. Developing fuel performance models requires fundamental understanding of potential failure mechanisms and how these mechanisms depend on the irradiation conditions and the material constituting the fuel. Accurate fuel performance modeling will also require good material properties and constitutive relations information.

Table 3 summarizes the key fuel failure mechanisms associated with TRISO particle fuel and how these mechanisms depend on reactor service conditions, particle design, and performance parameters. The failure mechanisms considered under irradiation are (1) pressure vessel failure; (2) cracking of the IPyC layer and IPyC layer partial debonding, leading to cracking of the SiC layer; (3) kernel migration; and (4) diffusive release through intact layers. Under hypothetical accident conditions, the failure mechanisms considered are (1) fission product attack of the SiC; (2) SiC thermal decomposition; (3) increase in SiC permeability/SiC degradation; (4) oxidation of the SiC layer; and (5) rapid energy deposition.

Table 4 summarizes the important material properties required for accurate modeling under irradiation and potential accident conditions and lists the state of knowledge of the specific properties, their importance to modeling, and potential measurement techniques. The ability to obtain measurements for all of these material properties is limited by program resources and, in some cases, by measurement science given the size of the TRISO particle, its individual constituents, and the nature of the actual measurement to be made.

The scope of this section is limited to activities needed to support fuel performance modeling. However, as indicated in Table 3, fission product release from the kernel and transport of fission products through the coating layers directly affect some failure mechanisms. The source term aspects of fission product transport behavior are covered under the Fission Product Transport and Source Term element of the program.

The R&D needs for fuel performance modeling are briefly summarized in the following subsections. The activities required to address these needs (fabrication of test articles, irradiation, and PIE) are addressed in the appropriate program element, with more detailed planning performed as the program proceeds.

Table 3. Summary of coated-particle failure mechanisms.

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Pressure vessel failure	Temperature Burnup Fast fluence	Strength of SiC Buffer density (void volume) Fission gas release CO production Particle asphericity Layer thicknesses Kernel type (UO_2, UCO)	
Irradiation-induced PyC failure	Fast fluence Temperature	Dimensional change of PyC Irradiation-induced creep of PyC Anisotropy of PyC Strength of PyC PyC thickness PyC density	
IPyC partial debonding	Temperature Fast fluence	Nature of the interface Interfacial strength Dimensional change of PyC Irradiation-induced creep of PyC	
Kernel migration	Temperature Burnup Temperature gradient	Layer thicknesses CO production Kernel type (UO_2 versus UCO)	Modeled with semi-empirical measured migration coefficient.
Diffusive release through intact layers	Temperature Burnup Temperature gradient Time at temperature	Chemical state/transport behavior of fission products Microstructure of SiC SiC thickness	Could be more important at high burnup in LEU fuels because of greater yields of Pd from Pu fissions and because of higher temperatures in future designs. More important under accident conditions.

Table 3. (continued).

Failure Mechanism	Reactor Service Conditions	Particle Design and Performance Parameters	Comments
Corrosion of SiC by CO	Temperature Burnup Time at temperature	Kernel type (UO_2, UCO) IPyC performance	CO is generated in particles with UO_2 kernels. At elevated temperatures, CO can attack the SiC layer if the IPyC layer is porous or has failed.
SiC thermal decomposition	Temperature Time at temperature	SiC thickness Microstructure of SiC	Not important in traditional accident envelope (peak temperature <1600°C). Expected to be important at ~2000°C. Degradation observed at 1800°C in coated particles was attributed to this mechanism but may have been fission product attack instead.
Increase in SiC permeability/SiC degradation	Burnup Temperature Fluence	Microstructure of SiC* Diffusion* Buffer densification and cracking* Thickness of SiC Permeability of SiC	Exact mechanism is unclear, but limited data from higher burnup fuel suggest increased fission product release under long-term heat-up. Could be fission product attack and would be more important at higher burnup in LEU fuels because of greater yields of Pd from Pu fission and higher operating and/or accident temperatures.
Oxidation of SiC layer	Partial pressure of oxygen Temperature Time at temperature	Thickness of SiC layer Microstructure of SiC layer	Results from external attack such as air or water. Needed for modeling kinetics of oxidation.
Rapid reactivity insertion	Energy deposition (J/g-fuel) Time duration of the deposition Burnup of fuel	Degree of kernel melting/vaporization Thickness of layers Coefficient of thermal expansion of layers Elastic modulus of layers Swelling of kernel Kernel-coating mechanical interaction	Limited data available. However, available data indicate that reactivity events in an HTGR are relatively benign in comparison to other technologies.
*Indicates a potential parameter			

Table 4. Key material properties needed for fuel performance modeling.

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Irradiation performance			
PyC anisotropy	Known to be critical to characterize PyC behavior. Ability to measure it accurately and precisely is needed.	All key properties are thought to depend on anisotropy.	Use x-ray, Raman laser, and optical methods.
PyC irradiation-induced dimensional change	Reasonably well known as a function of temperature and density. Key issue is link between shrinkage and anisotropy.	Stress depends on ratio of shrinkage rate to irradiation-induced creep.	Measure dimensional change on PyC specimens.
PyC irradiation-induced creep	Uncertain with a factor of 5, based on limited database. Would like to know creep as a function of temperature, density, and anisotropy.	Stress depends on ratio of shrinkage rate to irradiation-induced creep.	Use special specimens (split composite ring test).
Poisson's ratio in creep	Reasonably well known. Literature data range from 0.3 to 0.5. Best estimate is 0.4. Probably a function of density. Unclear whether it is a function of anisotropy.	Has modest effect on stress in PyC layer.	Use special specimens.
Strength of PyC	Data vary significantly. Some exist as a function of density and anisotropy. Key issue is how well the anisotropy of the PyC was known, because that determines the functional relationship.	Very important.	Obtain bistructural-isotropic-coated particles that can be tested using classic ring test or crush test.
Strength of SiC	Data vary significantly. Need data as a function of density, neutron fluence, irradiation temperature, and microstructure (large grain versus small grain and columnar versus equiaxial). Microstructure is a function of deposition conditions. Data are available for Chinese SiC. German data suggest that irradiation can reduce strength. The United States has correlated many data and concludes there is still uncertainty about effect of irradiation. There are non-trivial issues related to experimental procedures used in past measurements. The presence of free Si in the SiC layer can cause strength reductions.	Very important.	Can use irradiated particles as well as classic brittle ring technique. Also can use axial compression of a cylindrical plug inside SiC cylindrical sample. Key issue is linkage of data to microstructure.

Table 4. (continued).

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Interfacial bond strength between SiC and PyC	Very little is known. Historic value of ~50 MPa is used in calculations. Recent data that simulated SiC/PyC bond indicated strengths of 50 to 100 MPa. Tends to agree reasonably well with values from SiC/SiC composites.	Needed to understand the nature of debonding of the layers. The nature of the bond depends on the nature of the fabrication process.	Use special specimens and special punch/shear test to get bond strength.
Irradiation-induced swelling of SiC	Data are being obtained in United States fusion program. Swelling is on the order of 0.2 to 1.2% in temperature range of interest. More data in reactor-relevant temperature range (1000 to 1300°C) would be useful.	Lower importance given uncertainty in other parameters.	Take density (density gradient column) measurements.
Irradiation-induced SiC creep	Limited data at low fluence.	Modest impact. PyC creep is much larger effect.	Use split-ring or bend-strength relaxation techniques.
Fission gas release from the kernel	Data on gas release are reasonably well known for UO_2. Little to no data on UCO, especially at high burnup.	Direct contributor to pressure in particle.	Can be measured by crushing particles or online from "intentionally failed" particles.
CO production	Important for UO_2 fuel only. Data exist at low burnup from German program. No data at high burnup.	Direct contributor to pressure in particle and affects kernel migration.	Can be measured by crushing particles.
Kernel swelling	Reasonably well known at moderate burnup. More data at very high burnups would be useful.	Need to prevent kernel/ coating mechanical interaction.	Part of PIE planning for irradiated fuel.
Accident performance: long-term heating/air ingress/rapid reactivity transients			
Thermal expansion coefficient of PyC	Thermal expansion is different in the two orientations in PyC and depends on the anisotropy of the material. Effect of irradiation is not well known. Limited data available.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use conventional techniques. Small sample size adds to overall difficulty in measurement and uncertainty.

Table 4. (continued).

Property	Current State of Knowledge	Importance in Modeling	How to Measure
Elastic modulus of PyC	Modulus is a function of anisotropy, fluence, density, and temperature. Few to no data at very high temperatures expected in accidents.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use resonant ultrasound spectroscopy or nano-indentation.
Elastic modulus of SiC	Data from fusion program show a 10% drop at reactor-relevant temperatures and radiation doses. Little data above 1000°C.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use resonant ultrasound spectroscopy or nano-indentation.
Thermal expansion coefficient of SiC	Limited amount of data suggests expansion is constant between 900 and 1300°C. No systematic dependence on coating temperature or neutron irradiation. The presence of free carbon in SiC can reduce coefficient of thermal expansion by 40%.	Critical for potential reactivity events where large temperature gradients may develop within the fuel particle.	Use conventional techniques. Small sample size adds to overall difficulty in measurement and uncertainty.
Fission product interactions with layers and potential degradation of properties	Unknown influence at present.	Unknown at present.	Need to examine irradiated high-burnup particles that have been heated to determine magnitude of effect.
Buffer survivability	Failure of the buffer appears to be important to whether fission products get to the IPyC/SiC interface. This effect needs to be studied with the performance model before a definitive direction on the need for this work can be determined.	Have some properties on buffer strength and dimensional change to determine its failure; these can be used as a starting point for evaluations.	Need to produce some low-density material for material tests.
Kernel swelling under rapid energy deposition	Little data available under rapid energy deposition conditions for reactivity-induced accidents that are more severe than anticipated for HTGRs. For consideration under GIF HTGR fuel collaboration with Japan.	Kernel swelling and kernel-coating mechanical interaction may be critical to predicting failure in rapid reactivity transients.	Part of PIE following reactivity transient testing.

3.4.1 Thermomechanical and Thermophysical Properties of Coating Layers under Normal Operation

The thermomechanical and thermophysical properties of PyC and SiC listed in Table 4 are needed as a function of fast fluence and deposition conditions, where appropriate. In many cases, these measurements need to be made on samples of the individual materials because of the difficulty of making the measurement on the coated particle in situ. Examples of the properties include anisotropy of PyC, irradiation-induced dimensional change of PyC, irradiation-induced creep of PyC, PyC Poisson's ratio in creep, interfacial bond strength between SiC and PyC, irradiation-induced swelling of SiC, irradiation-induced creep of SiC, and Weibull strength of PyC and SiC. This work was initiated at ORNL and the University of Michigan but was halted in 2013 because of a lack of funding. No further work is planned at this time within the AGR Fuel program. A European program is under way as part of collaboration activities by the GIF VHTR Fuel and Fuel Cycle Program Management Board.

3.4.2 Thermochemical Properties of Kernel under Normal Operation

The thermochemical properties of the kernel listed in Table 4 are needed as a function of burnup. Fission gas release from UO_2 kernels is reasonably well understood. Fission gas release from UCO kernels is needed over the relevant burnup and temperature ranges for the HTGR. CO release from UO_2 kernels is also needed at burnups in excess of 10% FIMA at relevant reactor temperatures (up to 1300°C). Finally, measurements of kernel swelling for both UO_2 and UCO kernels are needed, especially at high burnup. These measurements will be made on UCO kernels as part of the AGR-3/4 experiment irradiation and associated PIE.

3.4.3 Thermomechanical and Thermophysical Properties of Coating Layers under Accident Conditions

Table 4 lists the properties needed to model the mechanical behavior of the coated particle under accident conditions. The thermal expansion coefficient and elastic modulus of PyC are needed as functions of fast fluence and temperature (1200 to 1800°C). Also needed are the corresponding properties of SiC. Work in these areas is not planned under the existing budget scenario. No proposed locations or personnel have been identified to perform this work should its priority increase.

3.4.4 Thermochemical Properties of Coating Layers under Accident Conditions

Fission products can interact with the SiC layer and degrade the properties of the layer. Of greatest concern is Pd attack under accident conditions. Many researchers have studied the attack of the SiC layer by Pd. The impact of the attack on the degradation of the properties of the layer has not been studied. Simple one-dimensional models assume that the particle fails when ~50% of the SiC layer has been attacked. A more sophisticated finite-element approach that models degradation and assesses the resulting thermomechanical response of the degraded coatings has been developed and is being implemented in the PARFUME code. Review of the historical data suggests that out-of-pile testing on ideal systems provides interaction rates that are orders of magnitude above that observed in coated particles. Measurements of fission product attack have been made during PIE of AGR-1 fuel compacts and will be made on AGR-2 fuel compacts.

Data from Germany suggest that the SiC layer becomes permeable to certain fission products under high-temperature heating when the coated particles are exposed to higher-burnup and fast-fluence conditions (14% FIMA, 6 to 8 × 10^{25} neutrons/m^2). The permeability may be associated with a microstructural change or corrosion of the SiC by CO above a critical concentration. Or the permeability may be a mischaracterization of the reason for the higher fission product releases because of uncertainties associated with the irradiation history (especially temperature) of the AVR pebbles that were tested. Further evaluation of the original data is needed.

Tests are planned to evaluate the oxidation behavior of SiC as part of the accident heat-up tests in AGR-5/6/7 in which the influence of air on fuel behavior will be studied. Low air partial pressures and fuel temperatures consistent with air-ingress calculations will be used.

Kernel swelling and kernel coating mechanical interaction may be critical for predicting failure in reactivity-insertion accidents. These data can be obtained as part of PIE following reactivity-insertion accident simulation testing. However, reactivity-insertion accident testing is not currently planned as part of the AGR Fuel program, because the likelihood of rapid (super prompt critical) reactivity transients that could induce fuel failures are precluded by the current prismatic HTGR design.

3.4.5 Thermophysical and Physiochemical Properties of Fuel Compacts

With the AGR fuel compacting process for HTGR fuel, thermophysical and physiochemical properties of the compact need to be measured to enable accurate fuel performance assessments in the HTGR irradiations. Of these properties, the irradiation-induced shrinkage and the thermal conductivity of the compact as a function of fluence and temperature need to be measured during PIE.

3.4.6 Code Benchmarking and Improvement

Currently, significant activity is taking place around the world to develop improved fuel performance codes under normal operating and potential accident conditions. The benchmarking of fuel performance codes took place under the auspices of the IAEA for both normal and potential accident conditions through 2008, based mainly on historical irradiations and safety tests.[51] Additional benchmarking is foreseen under the GIF/VHTR Fuel and Fuel Cycle Program Management Board based on the behavior of the current generation of T TRISO fuel in current irradiations and on safety tests planned performed in the United States and other international programs. INL has completed pre-test predictions for the AGR-1, AGR-2[52], and AGR-3/4[53] experiments. Safety test predictions have been completed for the AGR-1[54] and AGR-2[55] experiments; As-run analysis, safety test predictions, and fission product transport parameter estimation for AGR-3/4 will be performed in conjunction with AGR-3/4 PIE in FY 2018. Pre-test predictions and post-test calculations will be performed for the AGR-5/6/7 irradiation experiments. Similar sets of calculations will be performed for a subset of the safety tests using accident performance models, as determined by the AGR Fuel program. As the new material properties data in the earlier experiments become available, the calculations will be rerun to understand the influence of the improved data on predicted behavior. The performance test fuel, fuel qualification irradiations, and accident testing, along with planned material property irradiations (obtained via the DOE Nuclear Energy Research Initiative and international collaborations or by irradiation of material samples in HFIR at ORNL), will provide much of the separate-effects data needed to improve the fuel performance models.

3.5 Fission Product Transport and Source Term

The goal of the Fission Product Transport and Source Term activity was to produce a technical basis for source terms under normal and potential accident conditions for the HTGR. Initial studies were performed to measure hydrogen and tritium permeation into various high nickel superalloys that could potentially be used in an HTGR. Reports and papers were published[56,57] that discussed the outcome of these studies. However, work was halted in late 2011 because of the DOE-NE decision to defer further NGNP project work scope until a public-private partnership was firmly established. The recent announcement by DOE-NE of a Funding Opportunity Announcement award to X-energy LLC to lead a team that will pursue development of an HTGR is the first public-private partnership established in the United States, with ORNL and INL as team members also. As this award goes forward, further research may be performed in these areas. Under the INL ART TDO program office, further work scope regarding fission-product transport and source term has been cancelled for lack of a selected reactor design.

3.6 Other Activities

A few other activities in the AGR Fuel program are accounted for separately in the cost estimate in Section 4, because they do not fit easily into any one of the individual experiments or they cut across the different WBS elements in the program. These include:

- Reports that document the results of the AGR Fuel program at key times will be given to the HTGR engineering design and licensing organizations for developing topical reports or producing safety documentation for the proposed plant.
- Facilities at ORNL and INL have been upgraded, and more upgrades may be required in the future to accomplish irradiation and PIE activities. The experience to date has been that some of the infrastructure needed to carry out the AGR Fuel program was in need of repair/upgrade or did not exist. These upgrades and new capabilities have enabled the program to obtain the data outlined in the plan status update and path forward. A new safety heat-up test furnace for air/water/moisture ingress transient testing has to be designed, built, qualified for remote operation, and installed at INL prior to the start of AGR-5/6/7 PIE. The cost of this furnace is expected to be similar to the total cost of the FACS furnace at INL or the CCCTF furnace at ORNL.
- Upgrades to the Nuclear Data Management and Analysis System software used to qualify and store all of the data generated in the AGR Fuel program that incorporate the latest versions of underlying software and interfaces with the Internet are anticipated over the remaining life of the program.

4. PROGRAM SCHEDULE AND COST

A detailed activity-based schedule (life-cycle baseline) for the activities presented in this technical program plan for TRISO fuel has been developed and is used to guide and prioritize activities year by year. A higher-level summary of that schedule is shown in Figure 3. The critical path for the fuel qualification remains through the AGR-5/6/7 irradiation at this time and then shifts to PIE and safety testing once the irradiations are complete. Irradiation durations are determined by their location in ATR. AGR-1 (620 EFPDs) and AGR-2 (559 EFPDs) were longer irradiations because of the lower thermal flux in the respective large B irradiation positions. AGR-3/4 had a much shorter duration (369 EFPDs), because it was irradiated in the NEFT and was a fission product transport test rather than fuel qualification test. The AGR-5/6/7 irradiation will be approximately 500 to 550 EFPDs, because it will also be irradiated in the NEFT, a higher flux position in ATR. The AGR-5/6/7 irradiation is also a qualification and margin test for the final AGR-5/6/7 fuel. The durations for PIE and safety testing are based on (1) estimates of throughputs at ORNL and INL based on the scope of anticipated activities, considering historical and current experience at INL and ORNL for AGR-1 and AGR-2 PIE and safety testing; (2) anticipated learning-curve effects for the safety testing and PIE of later compacts; and (3) schedule overlaps in the safety testing and PIE-related activities for fuel from each of these compacts, with consideration of PIE and safety testing experience gained with the early test trains. Based on the project schedule shown in Figure 3, the fuel for the HTGR is anticipated to be qualified by FY 2024, assuming the funding levels required to accomplish the tasks are available. The FY 2016 revision of the

ATR Integrated Strategic Operations Plan dated October 27, 2016,[58] has revised the ATR core internals changeout reactor shutdown period to April 2020 through early January 2021.

A detailed cost breakdown is shown by year in Figure 4, Figure 5, and Figure 6. Fabrication, irradiation, and PIE and safety testing activities are grouped by experiment (AGR-1, AGR-2, etc.). Separate cost lines are shown for fuel performance modeling and fission product transport scopes. Additional lines are provided for the other activities described in Subsection 3.5 that cut across the program WBS elements. Costs in Figure 4, Figure 5, and Figure 6 are actual costs through FY 2015. The budget figures for FY 2016 are included, and life-cycle baseline estimates are provided for activities in FYs 2017 through 2024. In the figures, the costs are also broken down by each of the major activities in the WBS.

The AGR-5/6/7 experiment total life-cycle cost estimate is higher than the earlier AGR experiments for several reasons. First, the fuel fabrication costs are significantly greater because of the program decision made in February 2014 to fabricate new AGR-5/6/7 fuel kernels rather than use the original lot. The fuel kernels within the original lot met the fuel specifications but had fissures that were thought to fracture during the coating process and result in misshapen TRISO particle fuel. Thermal analysis of these misshapen particles demonstrated areas of excessive stress during irradiation were likely to occur and cause the particles to fail. Fabrication of new fuel kernels for the AGR-5/6/7 experiments required hiring of new operators and some staff at BWXT with related training and qualification. The kernel-fabrication equipment and processes had to be restarted, requiring maintenance and repair of equipment. This delay then caused a cascade of other delays—maintenance and repair to the coating, overcoating, and compacting equipment in order to be fully functional after an extended shutdown. Second, the AGR-5/6/7 test train design is much different than the previous experiment designs in order to accomplish the test objectives, which has increased the costs. Also, PIE performed to date on the AGR-2 experiment identified TC placement as having a possible negative effect on the TRISO fuel particle performance during irradiation. Third, the testing of moisture and air ingress on compacts during safety testing will be performed in a new furnace being developed. The development, fabrication, testing, and installation of the furnace in a suitable operating location will increase the costs associated with PIE and safety testing of the AGR-5/6/7 compacts. Fourth, the PIE and safety testing will be performed at both ORNL and INL in order to complete it in a timely manner. The costs related to performing this work at both laboratories increase as a result. A final impact to the overall costs of the AGR-5/6/7 experiments is the additional time that is being required to complete the entire testing as a result of the extended outages planned for ATR, the reduced number of annual EFPDs available for experiment irradiation, and the related need for management and oversight of operations over the longer timeframe.

The total program cost is estimated to be ~$367M, based on completing all activities described in this technical program plan, with no constraints put on annual funding levels. If the funding levels are constrained over this period, concessions will need to be made and priorities established as to which activities will be completed and which will be deferred or cancelled. PIE and safety testing, and fission product transport plans are based on certain assumptions with respect to the level of fuel performance and fission product transport model validation that the NRC will accept. If further examination and analysis are required above that planned, the schedule will be extended and costs will increase above those shown.

Fuel Development and Qualification

Activity	Start	Finish
AGR-1 - Shakedown Small Coater Fuel	**5/1/2003**	**9/15/2016**
AGR-1 Fuel Fabrication	5/1/2003	9/29/2006
AGR-1 Design and Assembly	5/1/2003	12/22/2006
AGR-1 Irradiation (Large B)	12/25/2006	11/6/2009
AGR-1 Test Train Cooldown	11/7/2009	2/28/2010
AGR-1 PIE and Safety Testing	3/1/2010	9/15/2016
AGR-2 Production Scale Coater Development	**12/2/2006**	**9/30/2022**
AGR-2 Fuel Fabrication	10/2/2006	4/9/2010
AGR-2 Design and Assembly	5/5/2008	6/11/2010
AGR-2 Irradiation (Large B)	6/23/2010	10/16/2013
AGR-2 Train Cooldown	10/17/2013	7/10/2014
AGR-2 PIE and Safety Testing	7/14/2014	9/30/2022
AGR-3/4 Fission Product Transport Data	**5/8/2006**	**9/30/2022**
AGR-3/4 Fuel Fabrication	5/3/2006	9/28/2011
AGR-3/4 Design and Assembly	3/2/2009	9/30/2011
AGR-3/4 Irradiation (NEFT)	12/14/2011	4/7/2014
AGR-3/4 Train Cooldown	4/7/2014	3/18/2015
AGR-3/4 PIE and Safety Testing	3/18/2015	9/30/2022
AGR-5/6/7 Fuel Qualification	**10/1/2008**	**9/27/2024**
AGR-5/6/7 Fuel Fabrication	10/1/2008	3/31/2017
AGR-5/6/7 Design and Assembly	5/1/2014	4/17/2017
AGR-5/6/7 Irradiation (NEFT)	10/15/2017	4/15/2021
AGR-5/6/7 Test Train Cooldown	4/15/2021	8/16/2021
AGR-5/6/7 PIE and Safety Testing	8/1/2021	4/15/2026
ATR Core Internals Changeout (no irradiation)	**4/19/2020**	**1/12/2021**
V&V of Fuel Performance Code	10/20/2017	12/30/2023
Fission Product Transport and Source Term Data Development	10/1/2003	TBD
Fission Product Transport and Source Term Code V&V	TBD	TBD

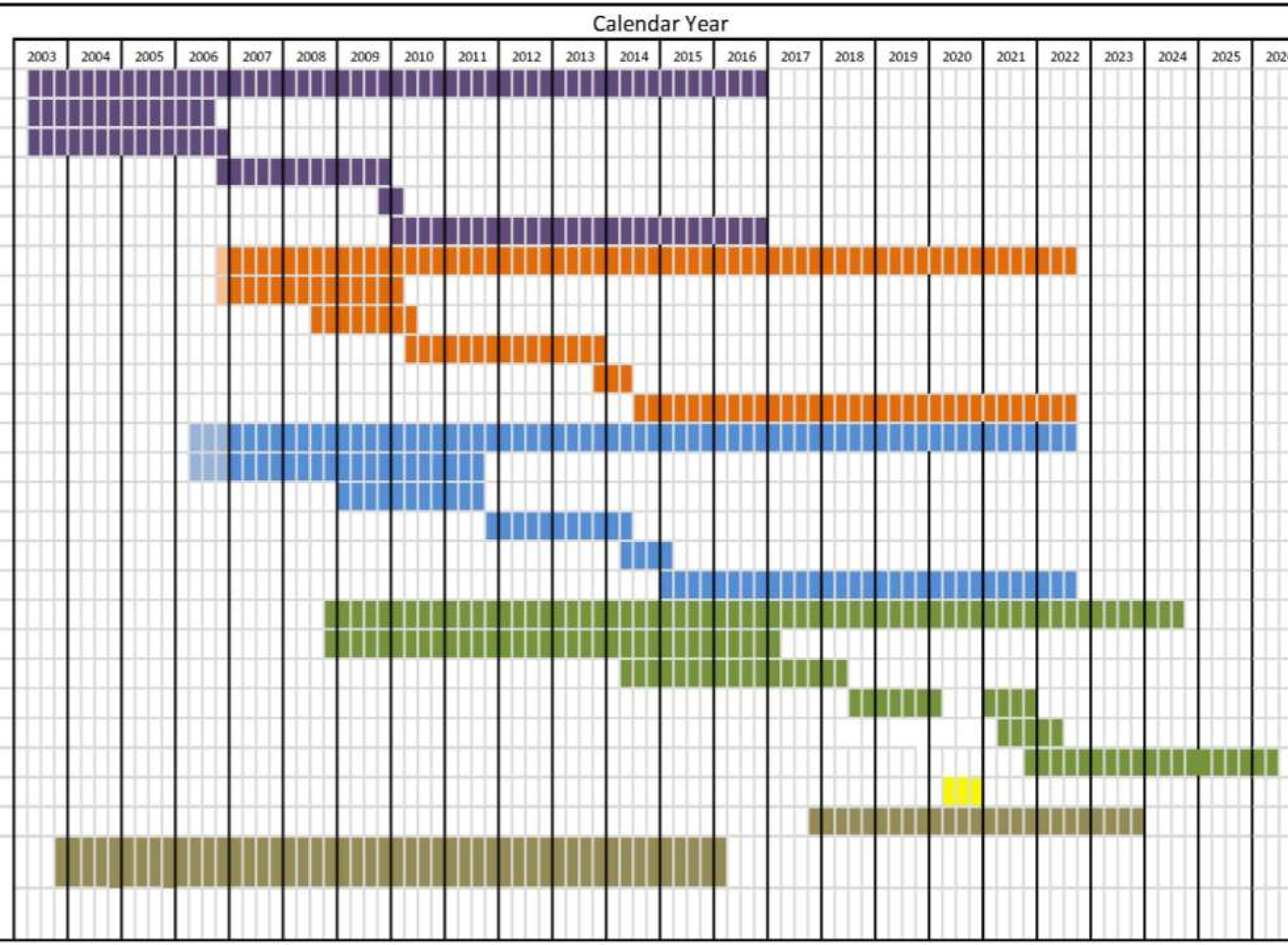

Figure 3. Fuel development and qualification higher-level summary schedule.

AGR-1 Shakedown Irradiation	FY-03	FY-04	FY-05	FY-06	FY-07	FY-08	FY-09	FY-10
Fuel Fabrication	$ 450	$ 3,185	$ 2,594	$ 4,602	$ 233	$ 35	$ 67	$ 166
Design and Assembly	$ 66	$ 696	$ 62	$ 1,370	$ 108	$ (29)		
Irradiation			$ 2,252	2110	$ 1,311	$ 1,832	$ 1,745	$ 1,425
PIE				$ 215	$ 248	$ 3,101	$ 7,252	$ 9,576
Data Qualification							$ 2,256	$ 1,617
TOTAL =	**$ 516**	**$ 3,881**	**$ 4,908**	**$ 8,297**	**$ 1,900**	**$ 4,939**	**$ 11,320**	**$ 12,784**
AGR-2 Production Scale Coater								
Fuel Fabrication					$ 2,110	$ 6,660	$ 2,102	$ 767
Design and Assembly						$ 212	$ 1,231	$ 2,018
Irradiation								$ 368
PIE								
Data Qualification								$ 337
TOTAL =	**$ -**	**$ -**	**$ -**	**$ -**	**$ 2,110**	**$ 6,872**	**$ 3,333**	**$ 3,491**
AGR-3/4 Fission Product Trans								
Fuel Fabrication					$ 350	$ 206	$ 187	$ 1,095
Design and Assembly			$ 685	120	$ 4	$ 5	$ 67	$ 617
Irradiation							$ 118	$ 534
PIE								
Data Qualification								
TOTAL =	**$ -**	**$ -**	**$ 685**	**$ 120**	**$ 354**	**$ 212**	**$ 372**	**$ 2,246**
AGR-5/6/7 Fuel Qualification								
Fuel Fabrication							$ 6,608	$ 6,333
Design and Assembly								
Irradiation								
PIE								
Data Qualification								
TOTAL =	**$ -**	**$ -**	**$ -**	**$ -**	**$ -**	**$ -**	**$ 6,608**	**$ 6,333**
Fuel Performance Modeling	$ 148	$ 371	$ 710	$ 620	$ 178	$ 661	$ 1,192	$ 1,256
Fission Product Transport		$ 82	$ 46	$ 71	$ 53	$ 396	$ 714	$ 736
NRC Reports								$ -
Fuel Fab Commercialization								$ -
Facility Upgrades						$ 2,309	$ 3,811	$ 1,527
NDMAS Upgrades								$ 1,545
PM Oversight	$ 592	$ 937	$ 1,077	$ 1,433	$ 645	$ 1,648	$ 1,331	$ 1,557
SUBTOTAL =	**$ 740**	**$ 1,389**	**$ 1,833**	**$ 2,124**	**$ 876**	**$ 5,014**	**$ 7,048**	**$ 6,621**
GRAND TOTAL =	**$ 1,256**	**$ 5,270**	**$ 7,426**	**$ 10,541**	**$ 5,240**	**$ 17,037**	**$ 28,682**	**$ 31,475**
Cumulative actual total	$ 1,256	$ 6,526	$ 13,952	$ 24,493	$ 29,733	$ 46,770	$ 75,451	$ 106,926
FY03-FY16 Total Actuals								
FY17 Estimated Costs								
FY18-FY24 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)								

Figure 4. Fuel development and qualification annual costs for FYs 2003 through 2010.

AGR-1 Shakedown Irradiation	**FY-11**	**FY-12**	**FY-13**	**FY-14**	**FY-15**	**FY-16**	**FY-17**	**FY-18**	**FY-19**
Fuel Fabrication	$ 102	$ 23							
Design and Assembly									
Irradiation	$ 2	$ 248	61						
PIE	$ 6,549	$ 5,165	$ 5,901	$ 4,936	$ 1,809	$ 1,636	$ 143		
Data Qualification	$ 254	$ 175	43	$ 80	$ 215	$ 144	$ 130		
TOTAL =	**$ 6,907**	**$ 5,611**	**$ 6,005**	**$ 5,016**	**$ 2,024**	**$ 1,780**	**$ 273**		
AGR-2 Production Scale Coater									
Fuel Fabrication									
Design and Assembly	$ 3								
Irradiation	$ 2,624	$ 1,263	$ 1,106	$ 743	$ 2				
PIE		$ 41	$ 305	$ 870	$ 2,832	$ 5,110	$ 4,550	$ 5,391	$ 4,500
Data Qualification	$ 1,053	$ 1,081	$ 212	$ 123	$ 279	$ 131	$ 110	$ 245	$ 246
TOTAL =	**$ 3,680**	**$ 2,385**	**$ 1,623**	**$ 1,736**	**$ 3,113**	**$ 5,241**	**$ 4,660**	**$ 5,636**	**$ 4,746**
AGR-3/4 Fission Product Trans									
Fuel Fabrication	$ 1,948	$ 246							
Design and Assembly	$ 3,499	$ 37							
Irradiation	$ 1,792	$ 2,757	$ 3,003	$ 2,468	$ 824				
PIE		$ 544	$ 583	$ 549	$ 1,487	$ 1,852	$ 2,650	$ 3,285	$ 3,299
Data Qualification	$ 91	$ 73	$ 607	$ 326	$ 450	$ 392	$ 300	$ 369	$ 370
TOTAL =	**$ 7,330**	**$ 3,657**	**$ 4,193**	**$ 3,343**	**$ 2,761**	**$ 2,244**	**$ 2,950**	**$ 3,654**	**$ 3,669**
AGR-5/6/7 Fuel Qualification									
Fuel Fabrication	$ 6,881	$ 4,558	$ 3,323	$ 2,910	$ 4,197	$ 3,833	$ 4,100	$ 500	
Design and Assembly			$ 466	$ 860	$ 2,280	$ 2,430	$ 2,450		
Irradiation					$ -	$ -	$ -	$ 1,500	$ 1,500
PIE					$ 31	$ 452	$ 1,900	$ 2,072	$ 2,080
Data Qualification					$ 6	$ 4	$ 25	$ 319	$ 320
TOTAL =	**$ 6,881**	**$ 4,558**	**$ 3,789**	**$ 3,770**	**$ 6,514**	**$ 6,719**	**$ 8,475**	**$ 4,391**	**$ 3,900**
Fuel Performance Modeling	$ 611	$ 758	$ 610	$ 455	$ 530	$ 1,029	$ 826	$ 945	$ 748
Fission Product Transport	$ 641	$ 174							
NRC Reports									
Fuel Fab Commercialization									
Facility Upgrades	$ 1,568	$ 836			$ 435	$ -			
NDMAS Upgrades	$ 1,353	$ 597	$ 262	$ 642	$ 659	$ 651	$ 950	$ 793	$ 796
PM Oversight	$ 1,639	$ 1,289	$ 736	$ 816	$ 1,567	$ 1,933	$ 1,984	$ 2,019	$ 2,029
SUBTOTAL =	**$ 5,812**	**$ 3,654**	**$ 1,608**	**$ 1,913**	**$ 3,191**	**$ 3,613**	**$ 3,760**	**$ 3,757**	**$ 3,573**
GRAND TOTAL =	**$ 30,610**	**$ 19,865**	**$ 17,218**	**$ 15,778**	**$ 17,603**	**$ 19,597**	**$ 20,118**	**$ 17,438**	**$ 15,888**
Cumulative actual total	$ 137,537	$ 157,401	$ 174,619	$ 190,397	$ 208,000	$ 227,597	$ 247,715		
FY03-FY16 Total Actuals									
FY17 Estimated Costs									
FY18-FY24 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)									

Figure 5. Fuel development and qualification annual costs for FYs 2011 through 2019.

AGR-1 Shakedown Irradiation	FY-20	FY-21	FY-22	FY-23	FY-24	FY-25	FY-26	FY-27	Total
Fuel Fabrication									$ 11,457
Design and Assembly									$ 2,273
Irradiation									$ 10,987
PIE									$ 46,531
Data Qualification									$ 4,914
TOTAL =									**$ 76,162**
AGR-2 Production Scale Coater									
Fuel Fabrication									$ 11,640
Design and Assembly									$ 3,464
Irradiation									$ 6,106
PIE	$ 4,500	$ 1,500							$ 29,599
Data Qualification	$ 247	$ 150							$ 4,214
TOTAL =	**$ 4,747**	**$ 1,650**							$ 55,023
AGR-3/4 Fission Product Trans									
Fuel Fabrication									$ 4,033
Design and Assembly									$ 5,034
Irradiation									$ 11,496
PIE	$ 3,218	$ 1,500							$ 18,967
Data Qualification	$ 372	$ 150							$ 3,500
TOTAL =	**$ 3,590**	**$ 1,650**							**$ 43,030**
AGR-5/6/7 Fuel Qualification									
Fuel Fabrication									$ 43,243
Design and Assembly									$ 8,486
Irradiation	$ 500	$ 1,800							$ 5,300
PIE	$ 2,071	$ 6,500	$ 9,460	$ 9,357	$ 4,303				$ 38,226
Data Qualification	$ 380	$ 411	$ 411	$ 409	$ 279				$ 2,564
TOTAL =	**$ 2,951**	**$ 8,711**	**$ 9,871**	**$ 9,766**	**$ 4,582**				**$ 91,211**
Fuel Performance Modeling	$ 1,147	$ 750	$ 1,548	$ 1,542	$ 1,731				$ 18,366
Fission Product Transport									$ 2,913
NRC Reports									$ -
Fuel Fab Commercialization									$ -
Facility Upgrades									$ 10,486
NDMAS Upgrades	$ 799	$ 796	$ 796	$ 793	$ 796				$ 12,228
PM Oversight	$ 2,035	$ 1,600	$ 1,600	$ 1,600	$ 1,600				$ 31,667
SUBTOTAL =	**$ 3,981**	**$ 3,146**	**$ 3,944**	**$ 3,935**	**$ 4,127**				**$ 75,659**
GRAND TOTAL =	**$ 15,269**	**$ 15,157**	**$ 13,815**	**$ 13,701**	**$ 8,709**	**$ -**	**$ -**	**$ -**	**$ 347,692**
Cumulative actual total									
FY03-FY16 Total Actuals									
FY17 Estimated Costs									
FY18-FY24 Projected Costs based on scheduled activities (Includes PM Oversight and Technical Integration)									

Figure 6. Fuel development and qualification annual costs for FYs 2020 through 2024.

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